NUREG–0847
Supplement No. 9

# Safety Evaluation Report
## related to the operation of Watts Bar Nuclear Plant, Units 1 and 2

Docket Nos. 50–390 and 50–391

Tennessee Valley Authority

U.S. Nuclear Regulatory Commission

**Office of Nuclear Reactor Regulation**

June 1992

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## ABSTRACT

This report supplements the Safety Evaluation Report (SER), NUREG-0847 (June 1982), Supplement No. 1 (September 1982), Supplement No. 2 (January 1984), Supplement No. 3 (January 1985), Supplement No. 4 (March 1985), Supplement No. 5 (November 1990), Supplement No. 6 (April 1991), Supplement No. 7 (September 1991), and Supplement No. 8 (January 1992) issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items, and proposed license conditions identified in the SER.

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APPENDICES

## ABBREVIATIONS

| | |
|---|---|
| AMSAC | anticipated transient without scram mitigation system actuation circuitry |
| ASME | American Society of Mechanical Engineers |
| CAP | corrective action program |
| CATD | Corrective Action Tracking Document |
| CNPP | Corporate Nuclear Performance Plan (NUREG-1232, Vol. 1) |
| CQC | complete quadratic combination |
| ECP | Employee Concerns Program |
| ECSP | Employee Concerns Special Program |
| EDG | emergency diesel generator |
| ERCW | emergency raw cooling water |
| FS | factor of safety |
| FSAR | final safety analysis report |
| HVAC | heating, ventilation, and air conditioning |
| IPEEE | independent plant examination for external events |
| LPZ | low population zone |
| MDOF | multi-degree of freedom |
| NRC | Nuclear Regulatory Commission |
| NRR | Office of Nuclear Reactor Regulation |
| PGP | procedure generation package |
| QA | quality assurance |
| RG | regulatory guide |
| SAT | systems approach to training |
| SER | safety evaluation report |
| SP | special program |
| SRP | Standard Review Plan |
| SSER | supplement to SER |
| TAC | technical assignment control |
| TER | technical evaluation report |
| TI | temporary instruction |
| TMI | Three Mile Island |
| TVA | Tennessee Valley Authority |
| 2DOF | two degrees of freedom |
| UHI | upper head injection |
| WBNPP | Watts Bar Nuclear Performance Plan (NUREG-1232, Vol. 4) |
| WISP | Work Information and Scheduling Program |

# 1 INTRODUCTION AND DISCUSSION

## 1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985), Supplement No. 4 (SSER 4, March 1985), Supplement No. 5 (SSER 5, November 1990), Supplement No. 6 (SSER 6, April 1991), Supplement No. 7 (SSER 7, September 1991), and Supplement No. 8 (SSER 8, January 1992). As of this date, the staff has completed review of the applicant's Final Safety Analysis Report (FSAR) up to Amendment 68.

The SER and SSERs were written in accordance with the format and scope outlined in the Standard Review Plan (SRP, NUREG-0800). Issues that arose as a result of the SRP review that were not closed out at the time the SER was published were classified into outstanding issues, confirmatory issues, and proposed license conditions (see Sections 1.7, 1.8, and 1.9, respectively, which follow).

In addition to the guidance of the SRP, the staff would from time to time issue generic requirements or recommendations in the form of bulletins and generic letters. Each of these bulletins and generic letters carries its own applicability, work scope, and acceptance criteria; some are applicable to Watts Bar. The implementation status was addressed in Section 1.14 of SSER 6. The staff is reevaluating the status of implementation of all bulletins and generic letters. Results of this reevaluation will be published in a future SSER.

Each of the following sections or appendices of this supplement is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to, and not in lieu of, the discussion in the SER unless otherwise noted. Accordingly, Appendix A is a continuation of the chronology of the safety review. Appendix B is an updated bibliography.* Appendix E is a list of principal contributors to this supplement. Appendix G, Errata, continues to make corrections on the SER and previous SSERs. Appendices C, D, F, and H through R are not changed by this SSER. Appendices S, T, U, V, W, X, and Y are added in this SSER.

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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*Availability of all material cited is described on the inside front cover of this report.

## 1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. Additional outstanding issues were added in SSERs that followed. This section updates the status of those items. The completion status of each of the issues is tabulated below with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date, status information is conveyed in the staff's summary of the monthly meeting regarding licensing status.

| Issue* | Status | Section |
|---|---|---|
| (1) Potential for liquefaction beneath ERCW pipelines and Class 1E electrical conduit | Resolved (SSER 3) | 2.5.4.4 |
| (2) Buckling loads on Class 2 and 3 supports | Resolved (SSER 4) | 3.9.3.4 |
| (3) Inservice pump and valve test program (TAC M74801) | Updated (SSER 5) | 3.9.6 |
| (4) Qualification of equipment | | |
| (a) Seismic (TAC M71919) | Resolved (SSER 9) | 3.10 |
| (b) Environmental (TAC M63591) | Under review (SER) | 3.11 |
| (5) Preservice inspection program (TAC M63627) | Under review (SER) | 5.2.4, 6.6 |
| (6) Pressure-temperature limits for Unit 2 | On hold | 5.3.2, 5.3.3 |
| (7) Model D-3 steam generator preheater tube degradation | Resolved (SSER 4) | 5.4.2.2 |
| (8) Branch Technical Position CSB 6-4 | Resolved (SSER 3) | 6.2.4 |
| (9) $H_2$ analysis review | Resolved (SSER 4) | 6.2.5 |
| (10) Safety valve sizing analysis (WCAP-7769) | Resolved (SSER 2) | 5.2.2 |
| (11) Compliance of proposed design change to the offsite power system to GDC 17 and 18 (TAC M63649) | Under review (SSER 3) | 8.2 |

*The TAC (technical assignment control) number that appears in parentheses after the issue title is an internal NRC control number by which the issue is managed through the Workload Information and Scheduling Program (WISP) and relevant documents are filed. Documents associated with each TAC number can be listed by the NRC document control system, NUDOCS/AD.

| Issue | Status | Section |
|---|---|---|
| (12) Fire protection program (TAC M63648) | Under review (SER) | 9.5.1 |
| (13) Quality classification of diesel generator auxiliary system piping and components (TAC M63638) | Resolved (SSER 5) | 9.5.4.1 |
| (14) Diesel generator auxiliary system design deficiencies (TAC M63638) | Resolved (SSER 5) | 9.5.4, 9.5.5, 9.5.7 |
| (15) Physical Security Plan (TAC M63657) | Under review (SER) | 13.6 |
| (16) Boron-dilution event | Resolved (SSER 4) | 15.2.4.4 |
| (17) QA Program (TAC M76972) | Updated (SSER 5) | 17 |
| (18) Seismic classification of cable trays and conduit (TAC R00508, R00516) | Resolved (SSER 8) | 3.2.1, 3.10 |
| (19) Seismic design concerns (TAC M79717, M80346): | | |
| (a) Number of OBE events | Resolved (SSER 8) | 3.7.3 |
| (b) 1.2 multi-mode factor | Resolved (SSER 9) | 3.7.3 |
| (c) Code usage | Resolved (SSER 8) | 3.7.3 |
| (d) Conduit damping values | Resolved (SSER 8) | 3.7.3 |
| (e) Worst case, critical case, bounding calculations | Under review (SSER 6) | 3.7.3 |
| (f) Mass eccentricities | Resolved (SSER 8) | 3.7.2.1.2 |
| (g) Comparison of set A versus set B response | Opened (SSER 6) | 3.7.2.12 |
| (h) Category 1(L) piping qualification | Resolved (SSER 8) | 3.9.3 |
| (i) Pressure relief devices | Resolved (SSER 7) | 3.9.3.3 |
| (j) Structural issues | Resolved (SSER 9) | 3.8 |
| (k) Update FSAR per 12/18/90 letter | Resolved (SSER 8) | 3.7 |
| (20) Mechanical systems and components (TAC M79718, M80345) | | |
| (a) Feedwater check valve slam | Under review (SSER 6) | 3.9.1 |
| (b) New support stiffness and deflection limits | Resolved (SSER 8) | 3.9.3.4 |
| (21) Removal of RTD bypass system (TAC M63599) | Resolved (SSER 8) | 4.4.3 |
| (22) Removal of upper head injection system (TAC M77195) | Resolved (SSER 7) | 6.3.1 |
| (23) Containment isolation using closed systems (TAC M63597) | Awaiting submittal (SSER 7) | 6.2.4 |

| Issue | Status | Section |
|---|---|---|
| (24) Main steam line break outside containment (TAC M63632) | Awaiting submittal (SSER 7) | 15.4.2 |
| (25) Health Physics Program (TAC M63647) | Under review (SSER 7) | 12.3, 12.5, 12.6, 12.7 |
| (26) Regulatory Guide 1.97, Instruments To Follow Course of Accident (TAC M77550) | Resolved (SSER 9) | 7.5.2 |
| (27) Containment sump screen design anomalies (TAC M77845) | Resolved (SSER 9) | 6.3.3 |
| (28) Emergency procedures (TAC M77861) | Resolved (SSER 9) | 13.5.2.1 |

## 1.8 Confirmatory Issues

SER Section 1.8 identified 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. This section updates the status of those items for which the confirmatory information has subsequently been provided by the applicant and for which review has been completed by the staff. The completion status of each of the issues is tabulated below, with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date, status information is conveyed in the staff's summary of the monthly meeting regarding licensing status.

| | | |
|---|---|---|
| (1) Design-basis groundwater level for the ERCW pipeline | Resolved (SSER 3) | 2.4.8 |
| (2) Material and geometric damping effect in SSI analysis | Resolved (SSER 3) | 2.5.4.2 |
| (3) Analysis of sheetpile walls | Resolved (SSER 3) | 2.5.4.2 |
| (4) Design differential settlement of piping and electrical components between rock-supported structures | Resolved (SSER 3) | 2.5.4.3 |
| (5) Upgrading ERCW system to seismic Category I (TAC M63617) | Resolved (SSER 5) | 3.2.1, 3.2.2 |
| (6) Seismic classification of structures, systems, and components important to safety (TAC M63618) | Resolved (SSER 5) | 3.2.1 |
| (7) Tornado-missile protection of diesel generator exhaust | Resolved (SSER 2) | 3.5.2, 9.5.4.1, 9.5.8 |
| (8) Steel containment building buckling research program | Resolved (SSER 3) | 3.8.1 |

| Issue | Status | Section |
|---|---|---|
| (9) Pipe support baseplate flexibility and its effects on anchor bolt loads (IE Bulletin 79-02) (TAC M63625) | Resolved (SSER 8) | 3.9.3.4 |
| (10) Thermal performance analysis | Resolved (SSER 2) | 4.2.2 |
| (11) Cladding collapse | Resolved (SSER 2) | 4.2.2 |
| (12) Fuel rod bowing evaluation | Resolved (SSER 2) | 4.2.3 |
| (13) Loose-parts monitoring system | Resolved (SSER 3) | 4.4.5 |
| (14) Installation of residual heat removal flow alarm | Resolved (SSER 5) | 5.4.3 |
| (15) Natural circulation tests (TAC M63603, M79317, M79318) | Under review (SER) | 5.4.3 |
| (16) Atmospheric dump valve testing | Resolved (SSER 2) | 5.4.3 |
| (17) Protection against damage to containment from external pressure | Resolved (SSER 3) | 6.2.1.1 |
| (18) Designation of containment isolation valves for main and auxiliary feedwater lines and feedwater bypass lines (TAC M63623) | Resolved (SSER 5) | 6.2.4 |
| (19) Compliance with GDC 51 | Resolved (SSER 4) | 6.2.7, App. H |
| (20) Insulation survey (sump debris) | Resolved (SSER 2) | 6.3.3 |
| (21) Safety system setpoint methodology | Resolved (SSER 4) | 7.1.3.1 |
| (22) Steam generator water level reference leg | Resolved (SSER 2) | 7.2.5.9 |
| (23) Containment sump level measurement | Resolved (SSER 2) | 7.3.2 |
| (24) IE Bulletin 80-06 | Resolved (SSER 3) | 7.3.5 |
| (25) Overpressure protection during low-temperature operation | Resolved (SSER 4) | 7.6.5 |
| (26) Availability of offsite circuits | Resolved (SSER 2) | 8.2.2.1 |
| (27) Non-safety loads powered from the Class 1E ac distribution system | Resolved (SSER 2) | 8.3.1.1 |
| (28) Low and/or degraded grid voltage condition (TAC M63649) | Updated (SSER 7) | 8.3.1.2 |

| Issue | Status | Section |
|---|---|---|
| (29) Diesel generator reliability qualification testing (TAC M63649) | Resolved (SSER 7) | 8.3.1.6 |
| (30) Diesel generator battery system | Resolved (SSER 2) | 8.3.2.4 |
| (31) Thermal overload protective bypass | Resolved (SSER 2) | 8.3.3.1.2 |
| (32) Update FSAR on sharing of dc and ac distribution systems (TAC M63649) | Under review (SSER 3) | 8.3.3.2.2 |
| (33) Sharing of raceway systems between units | Resolved (SSER 2) | 8.3.3.2 |
| (34) Testing Class 1E power systems | Resolved (SSER 2) | 8.3.3.5.2 |
| (35) Evaluation of penetration's capability to withstand failure of overcurrent protection device (TAC M63649) | Resolved (SSER 7) | 8.3.3.6 |
| (36) Missile protection for diesel generator vent line (TAC M63639) | Resolved (SSER 5) | 9.5.4.2 |
| (37) Component cooling booster pump relocation | Resolved (SSER 5) | 9.2.2 |
| (38) Electrical penetrations documentation (TAC M63648) | Under review (SER) | 9.5.1.3 |
| (39) Compliance with NUREG/CR-0660 (TAC M63639) | Resolved (SSER 5) | 9.5.4.1 |
| (40) No-load, low-load, and testing operations for diesel generator (TAC 63639) | Resolved (SSER 5) | 9.5.4.1 |
| (41) Initial test program | Resolved (SSER 3) | 14 |
| (42) Submergence of electrical equipment as result of a LOCA (TAC M63649) | Under review (SER) | 8.3.3.1.1 |
| (43) Safety parameter display system (TAC M73723, M73724) | Updated (SSER 6) | 18.2, App. P |

## 1.9 Proposed License Conditions

In Section 1.9 of the SER and SSERs, the staff identified 43 proposed license conditions. Since these documents were issued, the applicant has submitted additional information on some of these items, thereby removing the necessity to impose a condition. The completion status of the proposed license conditions is tabulated below, with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date, status information is conveyed in the staff's summary of the monthly meeting regarding licensing status.

| Proposed Condition | Status | Section |
|---|---|---|
| (1) Relief and safety valve testing (II.D.1) | Resolved (SSER 3) | 3.9.3.3, 5.2.2 |
| (2) Inservice testing of pumps and valves (TAC M74801) | Updated (SSER 5) | 3.9.6 |
| (3) Detectors for inadequate core cooling (II.F.2) (TAC M77132 and M77133) | Under review (SER) | 4.4.8 |
| (4) Inservice Inspection Program (TAC M76881) | Awaiting submittal (SSER 3) | 5.2.4, 6.6 |
| (5) Installation of reactor coolant vents (II.B.1) | Resolved (SSER 5) | 5.4.5 |
| (6) Accident monitoring instrumentation (II.F.1) | | |
| (a) Noble gas monitor (TAC M63645) | Resolved (SSER 5) | 11.7.1 |
| (b) Iodine particulate sampling (TAC M63645) | Resolved (SSER 6) | 11.7.1 |
| (c) High-range in-containment radiation monitor (TAC M63645) | Resolved (SSER 5) | 12.7.2 |
| (d) Containment pressure | Resolved (SSER 5) | 6.2.1 |
| (e) Containment water level | Resolved (SSER 5) | 6.2.1 |
| (f) Containment hydrogen | Resolved (SSER 5) | 6.2.5 |
| (7) Modification to chemical feedlines (TAC M63622) | Resolved (SSER 5) | 6.2.4 |
| (8) Containment isolation dependability (II.E.4.2) (TAC M63633) | Resolved (SSER 5) | 6.2.4 |
| (9) Hydrogen control measures (NUREG-0694, II.B.7) (TAC M77208) | Resolved (SSER 8) | 6.2.5, App. C |
| (10) Status monitoring system/BISI (TAC M77136, M77137) | Resolved (SSER 7) | 7.7.2 |
| (11) Installation of acoustic monitoring system (II.D.3) | Resolved (SSER 5) | 7.8.1 |
| (12) Diesel generator reliability qualification testing at normal operating temperature | Resolved (SSER 2) | 8.3.1.6 |
| (13) dc monitoring and annunciation (TAC M63649) | Under review (SSER 3) | 8.3.2.2 |
| (14) Possible sharing of dc control power to ac switchgear | Resolved (SSER 3) | 8.3.3.2.4 |

| Proposed Condition | Status | Section |
|---|---|---|
| (15) Testing of associated circuits | Resolved (SSER 3) | 8.3.3.3 |
| (16) Testing of non-Class 1E cables | Resolved (SSER 3) | 8.3.3.3 |
| (17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (II.G.1) (TAC M63649) | Resolved (SSER 7) | 8.3.3.4 |
| (18) Testing of reactor coolant pump breakers | Resolved (SSER 2) | 8.3.3.6 |
| (19) Postaccident sampling system (II.B.3) (TAC M77543) | Updated (SSER 5) | 9.3.2 |
| (20) Fire protection program (TAC M63648) | Under review (SER) | 9.5.1.8 |
| (21) Performance testing for communications systems (TAC M63637) | Resolved (SSER 5) | 9.5.2 |
| (22) Diesel generator reliability (NUREG/CR-0660) (TAC M63640) | Resolved (SSER 5) | 9.5.4.1 |
| (23) Secondary water chemistry monitoring and control program | Resolved (SSER 5) | 10.3.4 |
| (24) Primary coolant outside containment (III.D.1.1) (TAC M63646 and M77553) | Updated (SSER 6) | 11.7.2 |
| (25) Independent safety engineering group (I.B.1.2) (TAC M63592) | Resolved (SSER 8) | 13.4 |
| (26) Use of experienced personnel during startup (TAC M63592) | Resolved (SSER 8) | 13.1.3 |
| (27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2) (TAC M63656) | Awaiting submittal (SER) | 13.3 |
| (28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7) (TAC M77861) | Awaiting submittal (SER) | 13.5.2 |
| (29) Modifications to emergency operating instructions (I.C.8) (TAC M77861) | Awaiting submittal (SER) | 13.5.2 |
| (30) Report on outage of emergency core cooling system (II.K.3.17) | Resolved (SSER 3) | 13.5.3 |

| Proposed Condition | Status | Section |
|---|---|---|
| (31) Initial test program (TAC M79872) | Resolved (SSER 7) | 14.2 |
| (32) Effect of high-pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13) | Resolved (SSER 4) | 15.5.1 |
| (33) Voiding in the reactor coolant system (II.K.2.17) | Resolved (SSER 4) | 15.5.2 |
| (34) PORV isolation system (II.K.3.1, II.K.3.2) (TAC M63631) | Resolved (SSER 5) | 15.5.3 |
| (35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5) | Resolved (SSER 4) | 15.5.4 |
| (36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31) (TAC M77298) | Resolved (SSER 5) | 15.5.5 |
| (37) Detailed control room design review (I.D.1) (TAC M63655) | Updated (SSER 6) | 18.1 |
| (38) Physical Security Plan (TAC M63657) | Under review (SSER 1) | 13.6 |
| (39) Control of heavy loads (NUREG-0612) (TAC M77560) | Updated (SSER 3) | 9.1.4 |
| (40) Anticipated transients without scram (Generic Letter 83-28, Item 4.3) (TAC M64347) | Resolved (SSER 5) | 15.3.6 |
| (41) Steam generator tube rupture (TAC M77569) | Updated (SSER 5) | 15.4.3 |
| (42) Loose-parts monitoring system (TAC M77177) | Resolved (SSER 5) | 4.4.5 |
| (43) Safety parameter display system (TAC M73723 and M73724) | Opened (SSER 5) | 18.2 |

## 1.12 Approved Technical Issues for Incorporation in the License as Exemptions

The applicant applied for exemptions from certain provisions of the regulations. These have been reviewed by the staff and approved in appropriate sections of the SER and SSERs. These technical issues are listed below and the actual exemptions will be incorporated in the operating license:

(1) Seal leakage test instead of full-pressure test (Section 6.2.6, SSER 4) (TAC M63615)

(2) Criticality monitor (Section 9.1, SSER 5) (TAC M63615)

## 1.13 Implementation of Corrective Action Programs and Special Programs

On September 17, 1985, the NRC sent a letter to the applicant, pursuant to Title 10 of the Code of Federal Regulations, Section 50.54(f), requesting that the applicant submit information on its plans for correcting problems with the overall management of its nuclear program as well as on its plans for correcting plant-specific problems. In response to this letter, TVA prepared a Corporate Nuclear Performance Plan (CNPP) that identified and proposed corrections to problems with the overall management of its nuclear program, and a site-specific plan for Watts Bar entitled, "Watts Bar Nuclear Performance Plan" (WBNPP). The staff reviewed both plans and documented results in two safety evaluation reports, NUREG-1232, Vol. 1 (dated July 1987), and NUREG-1232, Vol. 4 (dated January 1990).

By letter dated September 6, 1991, the applicant submitted Revision 1 of the WBNPP. The staff has completed its review. Because Revision 0 was found acceptable, the review focused only on changes made in Revision 1 and if such changes made it necessary to review the staff's safety evaluation report (NUREG-1232, Vol. 4). The staff noted that most changes were made to reflect actions taken subsequent to submittal of Revision 0 and to reflect the current Watts Bar site organization and procedures. The staff recognizes that all the major changes documented in the WBNPP have been communicated in detail by TVA letters or in TVA/NRC meetings (and documented in subsequent meeting summaries), which are in turn tracked by specific NRR licensing actions or inspection issues. Thus, the staff concludes that Revision 1 of the WBNPP does not necessitate any revision of the staff's safety evaluation report, NUREG-1232, Vol. 4. The staff's efforts were tracked by TAC M81696 and M81697.

NUREG-1232, Vol. 4, documented the staff's general review of most of the corrective action programs (CAPs) and special programs (SPs) through which the applicant would effect corrective actions at Watts Bar. When the report was published, some of the CAPs and SPs were in their initial stages of implementation. The staff stated that it will report its review of the implementation of all CAPs and SPs and closeout of open issues in future supplements to the licensing SER, NUREG-0847. In accordance with that commitment, this new section was introduced in SSER 5 and will be updated in subsequent SSERs. The current status of all CAPs and SPs follows. The status described here fully supersedes that described in previous SSERs.

### 1.13.1 Corrective Action Programs

(1) Cable Issues (TAC M71917)

| | |
|---|---|
| Program review status: | Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), April 25, 1991 (the safety evaluation was reproduced in SSER 7 as Appendix P); supplemental safety evaluation dated April 24, 1992 (included as Appendix T in SSER 9). |
| Implementation status: | Full implementation expected by January 1994. |
| NRC inspections: | Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 |

(December 17, 1990); 50-390, 391/90-27 (December 20, 1990); 50-390, 391/90-30 (February 25, 1991); 50-390, 391/91-07 (May 31, 1991); 50-390, 391/91-09 (July 15, 1991); 50-390, 391/91-12 (July 12, 1991); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/92-01 (March 17, 1992); audit report of June 12, 1992 (Appendix Y of SSER 9); to come.

(2) Cable Tray and Tray Supports (TAC R00516)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 13, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.

Implementation status: Full implementation expected by September 1993.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); to come.

(3) Design Baseline and Verification Program (TAC M63594)

Program review status: Complete: Inspection Report 50-390, 391/89-12 (November 20, 1989); NUREG-1232, Vol. 4.

Implementation status: Full implementation expected by October 1993.

NRC inspections: Inspection Reports 50-390, 391/89-12 (November 20, 1989); 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20; (September 25, 1990); 50-390/91-201 (March 22, 1991); 50-390, 391/91-20 (October 8, 1991); 50-390, 391/91-25 (December 13, 1991); 50-390, 391/92-06 (April 3, 1992); to come.

(4) Electrical Conduit and Conduit Support (TAC R00508)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 1, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.

Implementation status: Full implementation expected by September 1993.

NRC inspections: Inspection Reports 50-390, 391/89-05 (May 25, 1989); 50-390, 391/89-07; (July 11, 1989); 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); to come.

(5) Electrical Issues (TAC M74502)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4.

| | |
|---|---|
| Implementation status: | Full implementation expected by December 1993. |
| NRC inspections: | Inspection Report 50-390, 391/90-30 (February 25, 1991); to come. |

(6) Equipment Seismic Qualification (TAC M71919)

| | |
|---|---|
| Program review status: | Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.10. |
| Implementation status: | Full implementation expected by June 1993. |
| NRC inspections: | Inspection Reports 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); audit report of May 14, 1992 (included as Appendix S of SSER 9); to come. |

(7) Fire Protection (TAC M63648)

| | |
|---|---|
| Program review status: | Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 9.5.1 of a future SSER. |
| Implementation status: | Full implementation expected by October 1993. |
| NRC inspections: | To come. |

(8) Hanger and Analysis Update Program (TAC R00512)

| | |
|---|---|
| Program review status: | Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 6, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3. |
| Implementation status: | Full implementation expected by July 1993. |
| NRC inspections: | Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-18 (September 20, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); audit report of May 14, 1992 (included as Appendix S of SSER 9); to come. |

(9) Heat Code Traceability (TAC M71920)

| | |
|---|---|
| Program review status: | Complete: Inspection Report 50-390, 391/89-09 (September 20, 1989); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 29, 1991. |

| | |
|---|---|
| Implementation status: | 100% (certified by letter, E. Wallace (TVA) to NRC, July 31, 1990); staff concurrence in SSER 7, Section 3.2.2. |
| NRC inspections: | Complete: Inspection Reports 50-390, 391/90-02 (March 15, 1990); 50-390, 391/89-09 (September 20, 1989). |

(10) Heating, Ventilation, and Air-Conditioning Duct and Duct Supports (TAC R00510)

| | |
|---|---|
| Program review status: | Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 24, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3. |
| Implementation status: | Full implementation expected by September 1993. |
| NRC inspections: | Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-01 (April 4, 1991); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (included as Appendix S of SSER 9); to come. |

(11) Instrument Lines (TAC M71918)

| | |
|---|---|
| Program review status: | Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 8, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), October 26, 1990 (the safety evaluation was reproduced as Appendix K in SSER 6). |
| Implementation status: | Full implementation expected by September 1993. |
| NRC inspections: | Inspection Reports 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-23 (November 19, 1990); 50-390, 391/91-02 (March 6, 1991); 50-390, 391/91-03 (April 15, 1991); 50-390, 391/91-26 (December 6, 1991); to come. |

(12) Prestart Test Program (TAC M71924)

| | |
|---|---|
| Program review status: | Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 17, 1989; NUREG-1232, Vol. 4; letter P. S. Tam (NRC) to D. A. Nauman (TVA), March 27, 1991. |
| Implementation status: | Withdrawn by letter (J. H. Garrity to NRC, February 13, 1992). Applicant will re-perform preoperational test program per Regulatory Guide 1.68, Revision 2. (See the staff's evaluation of FSAR Section 14 in a future SSER.) |

(13) Quality Assurance Records (TAC M71923)

| | |
|---|---|
| Program review status: | Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), December 8, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to M. C. Medford (TVA) June 9, 1992 (reproduced as Appendix X of SSER 9) |
| Implementation status: | Full implementation expected by February 1993. |
| NRC inspections: | Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-08 (September 13, 1990); 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-15 (September 5, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/92-05 (April 17, 1992); to come. |

(14) Q-List (TAC M63590)

| | |
|---|---|
| Program review status: | Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), January 23, 1991. |
| Implementation status: | Full implementation expected by January 1993. |
| NRC inspections: | Inspection Reports 50-390, 391/90-08 (September 13, 1990); 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/91-31 (January 13, 1992); to come. |

(15) Replacement Items Program (TAC M71922)

| | |
|---|---|
| Program review status: | Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), November 22, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 11, 1991 (the safety evaluation was reproduced as Appendix N in SSER 6); to come. |
| Implementation status: | Full implementation expected by November 1992. |
| NRC inspections: | Inspection Report 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); to come. |

(16) Seismic Analysis (TAC R00514)

| | |
|---|---|
| Program review status: | Complete: Letters, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7 and October 31, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.7. |
| Implementation status: | 100% (certified by letter, J. H. Garrity to NRC, December 2, 1991; staff concurrence in SSER 9, Section 3.7.1. |
| NRC inspections: | Complete: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); audit report by L. B. Marsh, October 10, 1990. |

(16)(a) Civil Calculation Program (TAC R00514)

A number of civil calculation categories are required by the Design Baseline and Verification Program CAP and constitute parts of the applicant's corrective actions. This program is regarded as complementary to but not part of the Seismic Analysis CAP. Staff efforts consist mainly of audits performed at the site and in the office (no program review).

| | |
|---|---|
| Implementation status: | Full implementation expected by June 1992. |
| NRC audits: | Memorandum (publicly available), T. M. Cheng to P. S. Tam, January 23, 1992; letter, P. S. Tam to D. A. Nauman (TVA), January 31, 1992; letter, P. S. Tam to M. O. Medford (TVA), May 26, 1992; to come. |

(17) Vendor Information Program (TAC M71921)

| | |
|---|---|
| Program review status: | Complete: Letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), September 11, 1990 (the safety evaluation was reproduced as Appendix I in SSER 5). |
| Implementation status: | Full implementation expected by March 1993. |
| NRC inspections: | Inspection Report 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); to come. |

(18) Welding (TAC M72106)

| | |
|---|---|
| Program review status: | Complete: Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 5, 1991. |
| Implementation status: | Full implementation expected by June 1992. |
| NRC inspections: | Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-18 (October 8, 1991); 50-390, 391/91-23 (November 21, 1991); to come. |

1.13.2 Special Programs

(1) Concrete Quality (TAC M63596)

| | |
|---|---|
| Program review status: | Complete: NUREG-1232, Vol. 4. |
| Implementation status: | Complete: Full implementation certified by letter, E. Wallace (TVA) to NRC, August 31, 1990; staff concurrence in SSER 7, Section 3.8.2.1. |
| NRC inspections: | Complete: NUREG-1232, Vol. 4; Inspection Reports 50-390, 391/89-200 (December 12, 1989); 50-390, 391/90-26 (January 8, 1991) |

(2) <u>Containment Cooling (TAC M77284)</u>

Program review status: Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), May 21, 1991 (the safety evaluation is reproduced as Section 6.2.2 of SSER 7).

Implementation status: Full implementation expected by February 1993.

NRC inspections: To come.

(3) <u>Detailed Control Room Design Review (TAC M63655)</u>

Program review status: Complete: NUREG-1232, Vol. 4; Section 18.1 and Appendix L of SSER 6.

Implementation status: Full implementation expected by April 1993.

NRC inspections: To come.

(4) <u>Environmental Qualification Program (TAC M63591)</u>

Program review status: NUREG-1232, Vol. 4; review in progress, results will be published in Section 3.11 of a future SSER.

Implementation status: Full implementation expected by August 1993.

NRC inspections: To come.

(5) <u>Master Fuse List (TAC M76973)</u>

Program review status: Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 6, 1991; letter, P. S. Tam to TVA Senior Vice President, March 30, 1992 (reproduced as Appendix U to SSER 9).

Implementation status: Full implementation expected by July 1992.

NRC inspections: Inspection Report 50-390, 391/86-24 (February 12, 1987); 50-390, 391/92-05 (April 17, 1992); to come.

(6) <u>Mechanical Equipment Qualification (TAC M76974)</u>

Program review status: NUREG-1232, Vol. 4; to come.

Implementation status: Full implementation expected by September 1993.

NRC inspections: To come.

(7) <u>Microbiologically Induced Corrosion (TAC M63650)</u>

Program review status: Complete: NUREG-1232, Vol. 4; Appendix Q of SSER 8.

Implementation status: Full implementation expected by January 1993.

NRC inspections: Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-13 (August 2, 1990); to come.

(8) Moderate Energy Line Break Flooding (TAC M63595)

Program review status: NUREG-1232, Vol. 4; to come.

Implementation status: Full implementation expected by May 1993.

NRC inspections: To come.

(9) Radiation Monitoring Program (TAC M76975)

Program review status: Complete: NUREG-1232, Vol. 4; this program covers areas addressed in Section 12 of the SER and SSERs.

Implementation status: Full implementation expected by December 1993.

NRC inspections: To come.

(10) Soil Liquefaction (TAC M77548)

Program review status: Complete: NUREG-1232, Vol. 4; letter, P. S. Tam to TVA Senior Vice President, March 19, 1992; Section 2.5 of SSER 9.

Implementation status: Full implementation expected by May 1992.

NRC inspections: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-23 (February 21, 1990); audit report by L. B. Marsh (October 10, 1990); audit report, P. S. Tam to D. A. Nauman, January 31, 1992; to come.

(11) Use-as-Is CAQs (TAC M77549)

Program review status: Complete: NUREG-1232, Vol. 4.

Implementation status: Full implementation expected by June 1992.

NRC inspections: Inspection Reports 50-390, 391/90-19 (October 15, 1990); 50-390, 391/91-08 (May 30, 1991); to come.

## 1.14 Implementation of Applicable Bulletin and Generic Letter Requirements

In SSER 5, Section 1.1, the staff stated that from time to time generic requirements or recommendations are issued in the form of bulletins and generic letters. The staff committed to prepare a summary of the implementation status of the applicable ones in SSER 6. The interim result of such effort was shown in Sections 1.14.1 and 1.14.2 of SSER 6. Because a long time has elapsed since these were addressed, the staff will reevaluate all bulletins and generic letters to determine if additional actions need to be taken. The staff will especially evaluate the appropriateness of implementation schedules. The evaluations will be completed before issuance of an operating license, and will be reported in a future SSER.

## 1.15 Employee Concerns Special Program

In NUREG-1232, Vol. 4 (Safety Evaluation Report on TVA: Watts Bar Nuclear Performance Plan, the staff stated on page 7-1 that it "has selected 15 technical issue subcategory reports for detailed evaluation and plans to examine portions of the other subcategory reports as part of the corrective action plan review."

The staff has reevaluated that commitment and determined that it has been met by review of all CAPs and SPs (see Section 1.13 of this SSER) for the following reasons: The 15 subcategory reports for Watts Bar were selected on the basis of significant issues as determined by the staff. That process would result in reviewing 15 of 107 subcategory reports whose issues constitute a subset of the 29 CAPs and SPs, and whose conclusions are at least two years older than those in the CAPs and SPs. The applicant's letter of July 13, 1989, titled "Corrective Action Program Plan Matrices," indicated that the Employee Concern Special Program (ECSP) findings and corrective actions formed part of the basis for the CAPs and SPs along with the additional information that has been learned since 1987. (The staff's evaluation is reported in Inspection Reports 50-390, 391/89-14, dated December 18, 1989, and 50-390, 391/90-05, dated May 10, 1990.) The staff's review and acceptance of the CAPs and SPs (see Section 1.13 of this SSER) indicated the staff's acceptance of the programmatic approaches to the resolution of the problems originated in the ECSP. The CAPs and SPs reflect current knowledge of the problems described in the ECSP and reflect the results from up to five additional years of hardware inspection and document review. On the other hand, the applicant has no plan to revise/update the subcategory reports. Therefore, the staff concludes that its commitment to review 15 subcategory reports has been obviated by its review of the 29 CAPs and SPs.

The applicant does update the ECSP on the basis of new findings by revising Employee Concern Corrective Action Tracking Document (CATD) implementation plans. Those updates ensure that the corrective actions continue to solve the problems identified before February 1, 1986. The CATDs referenced in the matrix of July 13, 1989, mirror the corrective actions identified in the CAPs and SPs. The implementation of the CATDs will be inspected by the staff in accordance with Temporary Instruction (TI) 2512/15. These inspections are continuing and will be completed before fuel is loaded into Unit 1.

This reevaluation was tracked by TAC M81696 and M81697.

# 2 SITE CHARACTERISTICS

## 2.1 Geography and Demography

### 2.1.3 Population Distribution

By Amendment 63 to the Final Safety Analysis Report, the applicant provided updated population data and projection based mainly on two sources: "County Population Estimates: July 1, 1987 and 1986," U.S. Bureau of the Census, and "County-Level Projections of Economic Growth and Population for Tennessee, Georgia and North Carolina," Bureau of Economic Analysis, U.S. Department of Commerce, 1986. The updated information leads the staff to revise certain findings originally published in the SER.

In Table 2.1 of the SER, the staff summarized the resident population in the Watts Bar vicinity. In response to more recent surveys and projections (mentioned above), the applicant submitted revised information in Amendment 63. Hence, Table 2.1 is reissued with the new information.

Table 2.1 Resident population in Watts Bar vicinity

| Year | 0-1 mi | 0-2 mi | 0-3 mi | 0-4 mi | 0-5 mi | 0-10 mi |
|---|---|---|---|---|---|---|
| 1970 | 35 | 190 | 540 | 1150 | 1750 | 10,735 |
| 1980 | 45 | 210 | 600 | 1295 | 2010 | 12,335 |
| 2020 | 23 | 363 | 976 | 2141 | 3833 | 19,223 |
| 2030 | 24 | 375 | 1008 | 2216 | 2964 | 19,855 |

The staff stated that the nearest densely populated center of 25,000 or more persons (as defined in 10 CFR Part 100) is the City of Oak Ridge, Tennessee, which had a 1980 population of 27,552, and whose nearest boundary is about 40 miles northwest of the Watts Bar site. The staff further stated that between the years 2000 and 2010, the population center is expected to shift to Athens, Tennessee, which is approximately 15 miles southeast of the site. Amendment 63 designates Cleveland, Tennessee, as the new population center, which had a 1990 population of 27,800. Cleveland is located approximately 30 miles south of the site. The new population center is well outside the low population zone (LPZ), which is defined by a circle of radius 3 miles from the site.

The applicant estimated that the population within a 50-mile radius of the site, based on data available in 1982, will grow to about 905,370 by 2020, representing a growth rate of 7.7 percent per decade. Using the new data, the applicant estimated that the population will increase to 1,099,647 by the year 2030, representing a growth rate of 9 percent per decade. The staff continues to believe that the applicant's projection of 9 percent per decade is conservative compared to the regional growth rate of 3.2 percent reported in the SER by the staff.

On the basis of this analysis, the staff concludes that the new information in Amendment 63 does not change the staff's conclusion conveyed in Section 2.1.4 of the SER. This review was tracked by TAC M77061.

## 2.5 Geology and Seismology

Since Supplement 3 to the Safety Evaluation Report (SSER 3) was issued in January 1985, the applicant submitted Amendments 54 through 63 to the FSAR. The staff reviewed these amendments and the applicant's letter, dated August 22, 1991, providing additional information. The staff's findings have been summarized in a letter to the applicant dated March 19, 1992. Following are details of the staff's findings.

### 2.5.4 Stability of Subsurface Materials and Foundations

In its August 22, 1991, letter, the applicant stated that Amendment 63 corrected certain typographical errors in FSAR Section 2.5.4.2.2.6.1. concerning the proportions of the different rock types. The applicant also stated that the information contained in this section is essentially unchanged since it was submitted prior to Amendment 54. The staff has accepted the information contained in that section in the SER and SSER 3. Similarly, the revisions made to FSAR section 2.5.4.8 dealing with liquefaction potential of soils were typographical clarifications that did not change the technical content of the FSAR. The staff has verified these facts.

The applicant stated that the revisions made by Amendments 54 through 63 to FSAR Section 2.5.4.2.2.9.2 dealing with the settlement of Category I structures were minor wording clarifications that did not change the technical content of the section. The staff previously accepted the applicant's evaluation of the effects of differential settlements on safety-related components and documented its acceptance in the SER and SSER 3.

The staff reviewed the calculations for the stability analyses of trench A at station 6+78 and trench B at station 2+50 of the underground barriers along the essential raw cooling water (ERCW) pipelines. These calculations were referenced in FSAR pages 2.5-147a through 2.5-147c. This review indicates that TVA did not provide any references to the sources for the values of the soil strength parameters used in its analyses. Furthermore, the calculation sheets show that the factor of safety (FS) for several cases is less than 1.0, whereas for other cases the FS exceeds 1.0. No explanation is provided in the calculation sheets for ignoring those cases that show an FS less than 1.0. During a telephone conference held on March 3, 1992, TVA staff stated that the stability analyses of these underground barriers were previously reviewed by the NRC staff before the issuance of SSER 3 in 1985 and again during an audit (publicly available audit report dated October 10, 1990), and found to be acceptable. However, the applicant agreed to review these calculations again, and provide proper explanation for the above questions regarding the stability analysis calculations. This is considered an open item. A similar open item related to the stability analysis of the intake channel slopes at Watts Bar is included in the audit report that was issued to TVA on January 31, 1992. These two open items will be treated as a single technical issue, since similar technical considerations govern both items, to be tracked as part of the Civil Calculations Program (see Item (16)(a), Section 1.13.1 of this SSER); the staff will report the resolution of this technical issue under the Civil Calculations Program.

Certain old sections of the FSAR (e.g., FSAR Sections 2.5.4.2.2.9, 2.5.4.2.2.9.1, and 2.5.4.2.2.9.2) have been eliminated, because the information previously given there has been incorporated in other sections under different headings. The staff has verified these facts.

In the last full paragraph on page 2-37 of the SER, the staff stated that "The computed factors of safety are listed in FSAR Table 362.28-4." The referenced Table 362.28-4 was deleted by Amendment 53 and its information was moved to Table 2.5-66 by Amendment 50. Other than the relocation, there is no change in technical content.

On the basis of a review of Amendments 54 through 63 to FSAR Sections 2.5.4 and 2.5.5, and TVA's response dated August 22, 1991, the staff determined that conclusions previously issued in the SER and SSER 3 remain unchanged. This review was tracked by TAC M77061.

# 3 DESIGN CRITERIA--STRUCTURE, COMPONENTS, EQUIPMENT, AND SYSTEMS

## 3.2 Classification of Structures, Systems, and Components

### 3.2.2 System Quality Group Classification

On page 3-3 of the SER, the staff stated that "The codes and standards used in the construction of TVA Classes A, B, C, D, G, or H components are identified in FSAR Tables 3.2-4, 3.2-5, and 3.2-6." The applicant has updated the FSAR so that this sentence should now read: "The codes and standards used in the construction of TVA Classes A, B, C, D, G, H, J, K, L, P, M, N, Q, S, U, or V components are identified in FSAR Tables 3.2-4, 3.2-5, and 3.2-6." The changes to Tables 3.2-4, 3.2-5, and 3.2-6 were communicated by Amendments 64 and 68 to the FSAR. The staff reported its review findings of Amendment 64 in SSER 6. The staff has reviewed information in Amendment 68 (TAC M82049 and 82050). The updated tables do not lead the staff to change conclusions already published in the SER and previous SSERs.

## 3.5 Missile Protection

### 3.5.1 Missile Selection and Description

#### 3.5.1.4 Missiles Generated by Natural Phenomena

In FSAR Amendments 57 and 64, the applicant identified a different tornado missile spectrum (Spectrum D) which was used for the design of the additional diesel generator building and additional Category I structures built after July 1979. In Section 3.5.1.4 of the SER, the staff concluded that the missile spectra (Spectra A, B, and C) used in design of all other (pre-July 1979) Category I structures were acceptable.

The new missile spectrum (Spectrum D) is equivalent to missile Spectrum II, identified in SRP Section 3.5.1.4. On the basis of its review of the new missile spectrum, the staff concludes that the spectrum meets the acceptance criteria of SRP Section 3.5.1.4 and conforms to the requirements of GDC 2 and 4 with respect to protection against natural phenomena and missiles. The new missile spectrum is, therefore, acceptable. The staff further concludes that the additional (fifth) diesel generator building and any other structure, system, or component designed to withstand missile Spectrum D is designed in accordance with the requirements of GDC 2 and 4 as they relate to protection against natural phenomena and externally generated missiles. This review was tracked by TAC M77061 and M77325.

## 3.7 Seismic Design

### 3.7.1 Seismic Analysis

In NUREG-1232, Volume 4, "Safety Evaluation Report on Tennessee Valley Authority: Watts Bar Nuclear Performance Plan," the staff stated that it will report the acceptability of TVA's implementation of the corrective action programs (CAPs) in the SSERs.

By letter dated December 2, 1991, TVA informed the staff that it has completed the Seismic Analysis CAP, thereby providing assurance that the seismic designs of structures, systems, and components are technically adequate and satisfy licensing requirements.

The staff reviewed the programmatic aspects of the Seismic Analysis CAP and found the CAP acceptable. The staff's review findings are documented in letters, S. C. Black to O. D. Kingsley, September 7 and October 31, 1989, and in NUREG-1232, Volume 4. The staff reviewed TVA's implementation in a team inspection and a site audit. Findings are documented in Inspection Report 50-390/89-21 (May 10, 1990) and in an audit report by L. B. Marsh, dated October 10, 1990 (publicly available).

On the basis of its reviews, inspection, and audit, the staff concurs with TVA that the Seismic Analysis CAP, as defined in TVA letters dated November 18, 1988, and June 29 and September 5, 1989, has been acceptably implemented for Unit 1.

### 3.7.3 Seismic Subsystem Design

In SSER 6, the staff identified Outstanding Issue 19(b) concerning the validity of the 1.2 multi-mode factor proposed by the applicant for use in the seismic evaluation of certain subsystems in the Watts Bar Nuclear Plant. The specific areas of application are: cable trays; conduits; and heating, ventilation, and air conditioning (HVAC) systems. The applicant's proposed factor of 1.2 is less than the 1.5 factor recommended in the Standard Review Plan (SRP) (NUREG-0800). The staff found that the applicant's original justification for the 1.2 factor was inadequate, in that the supporting calculations were too limited to accept for generic applicability to the systems proposed. The applicant responded by performing additional verification studies in support of the 1.2 multi-mode factor. The staff's evaluation of these studies, described in SSER 8, concluded that the additional verification studies had substantially improved the basis for the 1.2 factor. However, the staff found that the applicant's supporting calculations required additional confirmation. Specifically, the staff was concerned with the following two aspects of the applicant's supporting calculations: (1) the use of the CQC (complete quadratic combination) method in view of the RG 1.92 recommendations and (2) validity of the 2DOF (two degrees of freedom) predictions in terms of modal parameters. In a letter dated October 10, 1991, the applicant provided supplemental information to address these outstanding questions regarding the 1.2 multi-mode factor. The staff's evaluation of the October 10, 1991, submittal follows.

In addressing the staff's questions regarding the use of the CQC in conjunction with calculation WCG-1-397, the applicant has performed comparative studies comparing results from the CQC method with corresponding results from time-history analyses. The specific seismic models employed in these studies are: AA2150H, BC4150, CT13, and ELBTEE1. The latter represent different field configurations, including straight-run systems as well as systems with elbow and tee fittings.

Multi-mode effects were computed using the CQC as well as time-history analysis methods. These effects were quantified in terms of span force components as well as support forces from seismic inputs applied at the supports in different directions. Comparisons between the results from application of the CQC method and those from time-history analysis indicate that the time-history analyses produced, in some cases, higher ratios than the CQC method. For example, by comparing Tables A and B of the applicant's October 10, 1991, submittal, it can be seen that the multi-mode ratios associated with support forces for systems AA2150H, BC4150 and CT13 are generally higher when computed through time-history analysis than using the CQC method. A similar behavior is also observed, to a lesser extent, for system ELBTEE1 by comparing Tables G1 and D1 as well as C2 and D2 for x and z excitations, respectively. Although this study does not confirm the conservatism of the CQC method compared to time-history analysis, it did confirm that the applicant's proposed use of the 1.2 multi-mode factor is still conservative compared to the time-history results. Since the time-history approach is an acceptable method of analysis in the SRP, the staff considers its use an acceptable method to demonstrate the adequacy of the applicant's proposed multi-mode factor. On this basis, the applicant's time-history analysis resolves the staff's concern regarding the use of the CQC method to justify the multi-mode factor.

With respect to the second question related to the 2DOF frequency predictions, a key element in the applicant's proposed approach is that the 1.2 factor is applied to the highest spectral acceleration based on a frequency that is equal to or greater than that obtained from the 2DOF model. Furthermore, the applicant argued that the 2DOF model yields lower bound frequencies. The applicant then concluded that this approach would produce conservative results. The applicant had provided comparisons of the fundamental frequencies between the models analyzed using GTSTRUDL and the 2DOF approach. Although the fundamental frequencies agreed reasonably well, the staff asked the applicant to provide comparisons of the mode shapes to confirm the validity of the 2DOF model.

In response to the staff's questions regarding the validity of the 2DOF predictions of the modal parameters for the systems of interest, the applicant performed a set of comparative studies using multi-degree-of-freedom (MDOF) models. In these studies, the following five plant configurations were considered: BC4150, CT13, AA2150H, ELBTEE1, and ELBTEE2. Using these systems, the applicant, in the October 10, 1991, submittal, has compared their modal parameters using both detailed MDOF as well as simplified 2DOF calculations. For some of the above systems, for example, BC4150, CT13, and ELBTEE1, the 2DOF approach yields frequencies less than the MDOF frequency which corresponded to the second mode. The staff finds these results to be an acceptable demonstration of the validity of the 2DOF approach since the first mode is virtually a zero contributing mode. For the remainder of the systems considered, the 2DOF simplified predictions of system frequency were higher than the second modal frequency obtained through MDOF calculations. A closer examination of the applicant's results reveals that for these particular systems the second mode was not the dominant mode. For example, the 2DOF approach as applied to the system AA2150H produced a frequency that was higher than the second mode frequency of a detailed MDOF model for the same system. It is shown, however, that the dominant mode for the particular span under consideration is the third mode, which has higher modal frequency than that predicted by the 2DOF approach. In this sense, the 2DOF approach yields low-bound frequency estimates. Finally, to further support the validity of the proposed approach, the applicant also calculated the total response using all modes of interest; this is shown to be less than that computed using the 1.2 multi-mode factor. On this basis, the staff's concern regarding the validity of the 2DOF model predictions has been adequately addressed.

On the basis of its evaluation, the staff concludes that the applicant's October 10, 1991, submittal has adequately addressed the concerns expressed in SSER 8 regarding the validity of the 1.2 multi-mode factor. Therefore, Outstanding Issue 19(b) is resolved.

## 3.8 Design of Category I Structures

Outstanding Issue 19(j) was opened in SSER 6. By letter dated May 8, 1991, the applicant supplied information for resolving the items covered by Outsanding Issue 19(j). Most of the open items required simple clarifications that the staff reviewed and found acceptable. Items that required significant staff review are discussed below. Additional information was provided on the issue of thermal stresses by the applicant's letters dated June 6, August 22, October 16, and November 27, 1991.

The use of ductility ratio and associated specific limits was proposed for structural steel design in the load combination; in particular, in association with thermal load from accidents (Ta). The upper limit of 1.3 ductility ratio as proposed in TVA's November 27, 1991, letter implies that strain or displacement of the structure member may reach up to 30 percent beyond elastic limits. In TVA's letter, various other limits are proposed, mostly lower than 1.3 depending on the state of stresses or loading conditions. Some members (defined as secondary members) are allowed a ratio of 10.

The steel structures of the type at issue here are designed to remain within elastic limits. NRC made an exception to the design of impact barriers (Appendix A of SRP Section 3.5.3) where some ductility is allowed. In its response to the staff's request for additional information, the applicant indicated that the proposed approach does not have any parallel in any other industry applications. There is considerable strain capacity inherent in a ductile steel material after the material reaches its yield point. However, for structural members subjected to axial and lateral loads, the proposed criteria can lead to substantial reduction in margins of safety. Reasons for the applicant's original reluctance to go beyond elastic limit are insufficient experimental data for ductility-associated failure and lack of reliable analytical tools to calculate such ductility.

In a meeting on October 31, 1991, the staff formally communicated its concerns (see meeting summary dated November 8, 1991). The staff was concerned about the lack of experimental data associated with the proposed criteria, not only for the data that relate structural collapse (imminent failure) at the level of the proposed ductility ratio, but also the lack of data to verify the calculational methodology in an inelastic region. The calculational methodology is important since all the ductility ratios of the structures are obtained by calculations (computer code ANSYS), including seismic responses, in an inelastic region. It is the staff's position that any inelastic-nonlinear analysis should be backed up by physical data as well as by rigorous numerical analyses (such as error and stability). Lack of data in this regard has been identified by the applicant's own exhaustive literature and data search, as well as by consultation with experts in the field (see applicant's November 27, 1991, letter).

The applicant has presented much discussion about the Howland and Newmark test (June 6, 1991, letter). Though this is a valid example of verifying a code such as ANSYS, there should be a wider inclusion of data in such verification including dynamic responses, in an inelastic region. The applicant's discussion regarding elastic ratchet of dynamic load beyond elastic limit should not be based solely on sample calculations. The rest of the discussions regarding various issues such as beam-column and torsional buckling are also based on calculations without test data.

In the November 8, 1991, letter, the staff requested information regarding analytical verification of the computer code ANSYS with regard to numerical studies for error estimates and instability associated with the calculations. The applicant stated that the code was verified by means of various test runs of special finite element components. This type of verification cannot substitute for a rigorous numerical analysis of error and stability in an inelastic region.

The staff stated, in its January 27, 1992, letter, that, for the reasons stated above, the applicant should use the provisions of SRP Section 3.8.4 regarding thermally induced stresses which do not rely on ductility ratio. The applicant responded in its April 6, 1992, letter that a methodology consistent with the SRP will be used for Watts Bar for the design of steel members that are subjected to thermal restraints. It further stated that Watts Bar will use the linear elastic provisions of Design Guide DG-C 1.6.12, Revision 1, entitled "Evaluation of Steel Structure With Thermal Restraint," except for energy balance provisions of Section C.2.3.1. This commitment is acceptable to the staff.

The staff expressed a concern regarding the use of Revision 0 of NCIG-02 in assessing the safety-related weld at Watts Bar. The staff has approved Revision 2, not Revision 0. The applicant stated in its May 8, 1991, letter that any further sampling reinspection of structural welds after the issuance of NCIG-2, Revision 2 will be performed in accordance with NCIG-2, Revision 2 requirements. This commitment is acceptable to the staff.

The above evaluation fully resolves Outstanding Issue 19(j).

## 3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

In SSER 8, the staff stated that Outstanding Issue 4(a) will be considered fully resolved when Item 3.10.1(1), which was last updated in SSER 6, is resolved. The staff performed an audit between February 24 and 28, 1992, and documented its findings in an audit report (see Appendix S in this SSER). The audit report resolved Item 3.10.1(1). Hence Outstanding Issue 4(a) is fully resolved.

# 5 REACTOR COOLANT AND CONNECTED SYSTEMS

## 5.2 Integrity of Reactor Coolant Pressure Boundary

### 5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The staff has reviewed FSAR Section 5.2.7, as revised up to Amendment 67. In Section 5.2.5 of the SER, the staff identified and evaluated the upper head injection (UHI) system as a possible intersystem reactor coolant leakage path. The UHI system has been eliminated from the Watts Bar design (see Section 6.3.1.1 of SSER 7) and, therefore, the discussion of the UHI system in SER Section 5.2.5 no longer applies. However, this does not affect the original evaluation and the conclusions reached in Section 5.2.5 of the SER are still valid. This review was tracked by TAC M81221 and M81222.

# 6 ENGINEERED SAFETY FEATURES

## 6.3 Emergency Core Cooling System

### 6.3.3 Testing

In the Watts Bar SER (NUREG-0847, dated June 1982), the staff approved the proposed sump design in the FSAR based on the 1:4 scale model test performed by Norris Laboratory in March 1979 (Norris Lab Report WM28-1-85-101, docketed by letter to the NRC dated May 23, 1979).

A deviation was identified during an NRC inspection conducted between July 21 and August 20, 1986 (see Inspection Report 50-390/86-18 and 50-391/86-18). This deviation consisted of a discrepancy between the actual installation and the proposed FSAR design. The actual installation includes a 1/4-inch mesh screen attached to the outside of the trash rack; however, the proposed design includes a 1/2-inch mesh screen attached to the inside of the trash rack. This anomaly was tracked as Outstanding Issue 27 in SSER 7.

TVA responded to the deviation in a letter dated November 26, 1986, and stated that it would revise the FSAR. TVA explained that the 1/4-inch mesh screen was based on the results of the scale model test documented in the March 1979 Norris report. During a subsequent NRC inspection conducted between October 18 and November 20, 1989 (Inspection Report 50-390, 391/89-20), the inspector raised some questions about the validity of the scale model test because of the incorrect location of the 1/4-inch-mesh screens. Specifically, the inspector asked whether the different location and mesh size of the screen would cause a change in vortexing for the pump, and if the outside location of the screen would negatively affect the structural integrity of the sump if it became clogged. TVA contracted with Norris to reevaluate the test data to reflect the actual sump installation, and determine if there were any differences in its original conclusions. This reevaluation was documented in Norris Lab Report WR28-2-85-131, dated July 1989 (docketed by letter to NRC dated April 13, 1992). The inspector reviewed this report and concurred with the results, but stated that the Office of Nuclear Reactor Regulation (NRR) should approve the design change. On December 18, 1991, the NRR staff visited Watts Bar to inspect the sump and review the Norris report which was then not docketed.

On the basis of its review of the July 1989 Norris report, the staff concluded that the as-installed sump screen is an acceptable design. Therefore, Outstanding Issue 27 is considered resolved.

# 7 INSTRUMENTATION AND CONTROLS

## 7.5 Safety-Related Display Information

### 7.5.2 Postaccident Monitoring System

In the SER, the staff stated that the applicant did not use Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," for the design because the design preceded the publication and implementation dates of the guide. In SSER 7, Outstanding Issue 26 was opened to track resolution of this issue.

Generic Letter 82-33 (Supplement 1 to NUREG-0737) asked utilities to report to the staff describing how the postaccident monitoring instrumentation meets the guidelines of RG 1.97 as applied to emergency response facilities. After the generic letter was issued, the staff held regional meetings in February and March 1983 to answer licensee and applicant questions and concerns regarding the NRC policy on RG 1.97. At these meetings, it was established that the NRC review would only address exceptions taken to the guidance of RG 1.97. Further, where licensees or applicants explicitly state that instrument systems conform to provisions of the regulatory guide, no further staff review would be necessary for those items. Therefore, the review only addresses exceptions to the guidance of RG 1.97.

The applicant responded to Item 6.2 of the generic letter on January 30, 1984. Additional information was provided in a letter dated April 16, 1985. These submittals were replaced by a letter dated August 31, 1990, which was supplemented by letters dated October 11, 1990, and January 3 and October 29, 1991.

A detailed review and technical evaluation of the applicant's submittals was performed by EG&G Idaho, Inc., under a contract to the staff. EG&G reported this work in Technical Evaluation Report (TER) EGG-NTA-9563, "Conformance to Regulatory Guide 1.97 [Revision 2]: Watts Bar-1/-2," dated February 1992 (Appendix V of this SSER). The staff has reviewed this report and concurs with its conclusion that the applicant either conforms to, or has adequately justified deviations from, the guidance of RG 1.97 for each postaccident monitoring variable. Thus, Outstanding Issue 26 is considered resolved.

## 7.7 Control System Not Required for Safety

### 7.7.8 Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC)

By letter dated December 28, 1989, the staff informed the applicant that it has completed review of submittals concerning the proposed plant-specific AMSAC design at Watts Bar. The staff concluded that, pending the final resolution of the technical specification issue, the Watts Bar AMSAC design is in compliance with 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." The safety evaluation, performed under TAC M74501, is included as Appendix W in this SSER.

The staff reviewed Section 7.7.1.12 of the FSAR as revised up to Amendment 65, and concludes that the information therein is in accordance with the safety evaluation (Appendix W). This review was tracked by TAC M80143 and M80144.

# 9 AUXILIARY SYSTEMS

## 9.2 Water Systems

### 9.2.1 Essential Raw Cooling Water and Raw Cooling Water Systems

In Section 9.2.1 of the SER, the staff indicated that the essential raw cooling water (ERCW) system supplied cooling water to room coolers in the instrument rooms in addition to other identified loads. In Amendment 65 to the FSAR, the applicant stated that instrument room chillers would be used in lieu of room coolers supplied by the ERCW system. Therefore, the ERCW system, instead of room coolers, provides cooling to the instrument room chillers. The staff is identifying this change for clarification purposes only and, therefore, the evaluation and conclusion reached in the SER and its supplements are still valid. This review was tracked by TAC M80143 and M80144.

### 9.2.4 Potable and Sanitary Water Systems

In Section 9.2.4. of the SER the staff incorrectly noted the daily potable water requirements for Watts Bar as 7500 gallons per day (gpd). The actual estimated average potable water requirement is about 60,000 gpd with a maximum demand of approximately 80,000 gpd. The daily water demand requirement for the potable water system is not a consideration for the staff's acceptance of the potable water system, and was given for informational purpose only. Therefore, the original evaluation and conclusion of the SER remain unchanged. This review was tracked by TAC M80451 and M80452.

## 9.4 Heating, Ventilation, and Air Conditioning System

### 9.4.1 Control Room Area Ventilation System

In Section 9.4.1 of the SER, the staff described and evaluated chlorine detectors located in the control building outside air intakes. In Section 6.4 of SSER 5, the staff evaluated and found acceptable the removal of the chlorine detectors because small quantities of chlorine were stored on site and negligible amounts were stored off site in close proximity. The staff reviewed the FSAR up to Amendment 66, and hereby updates this section to conform with Section 6.4 of SSER 5. For clarification, the control room area automatically isolates upon the actuation of a safety injection signal from either unit, or upon detection of high radiation or smoke concentrations in the outside air supply stream.

This information is being given to clarify the design of the control building ventilation system. The conclusion reached in the SER and its first eight supplements is still valid. This review was tracked by TAC M80451 and M80452.

### 9.4.5 Engineered Safety Features Ventilation System

In Section 9.4.5 of the SER, the staff evaluated the diesel generator ventilation system and concluded it was acceptable. Since then, the applicant has added one more diesel generator and its support systems. The additional diesel generator building ventilation system has the same design bases as the diesel generator building ventilation system evaluated in the SER. The staff has reviewed the design of the additional diesel generator building ventilation system (FSAR as revised up to Amendment 66) and concludes that the evaluation and conclusion in the SER are also applicable to the additional diesel generator building ventilation system. The design is, therefore, acceptable. This review was tracked by TAC M80451 and M80452.

## 9.5 <u>Other Auxiliary Systems</u>

### 9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

#### 9.5.4.1 Emergency Diesel Engine Auxiliary Support Systems (General)

In Sections 9.5.4.1 through 9.5.8 of the SER, SSER 3, and SSER 5, the staff evaluated the emergency diesel generator (EDG) auxiliary support systems. These systems consist of the following:

- fuel oil storage and transfer system (Section 9.5.4.2)
- cooling water system (Section 9.5.5)
- starting system (Section 9.5.6)
- lubrication system (Section 9.5.7)
- combustion intake and exhaust system (Section 9.5.8)

The system description and evaluation in the SER and its supplements address the original four EDGs. A fifth EDG has been added in a separate building. The additional diesel generator and its auxiliary support system are essentially the same design as the original 4 diesel generators and their support systems. The evaluation and conclusion reached by the staff regarding the EDG auxiliary support systems in the SER, SSER 3 and SSER 5 also apply to the additional EDG. The staff reviewed the FSAR up to Amendment 65 and concludes that the auxiliary support systems for the additional EDG are, therefore, acceptable. This review was tracked by TAC M80143 and M80144.

#### 9.5.4.2 Emergency Diesel Fuel Oil Storage and Transfer System

In Section 9.5.4.2 of the SER the staff inferred that there were four embedded fuel oil storage tanks with a total capacity of 68,000 gallons. The FSAR as revised up to Amendment 65 states that there are a total of five embedded storage tanks, each with a capacity of 68,000 gallons to provide 7 days of operation for each EDG. This provides updated information but does not alter the staff's previous conclusion. This review was tracked by TAC M80143 and M80144.

# 10 STEAM AND POWER CONVERSION SYSTEM

## 10.4 Other Features

### 10.4.1 Main Condenser

In Section 10.4.1 of the SER, the staff described the main condenser as having three shells. As depicted in the FSAR revised up to Amendment 63, the main condenser is actually a single-shell, triple-pressure type with a divided waterbox. This clarification does not affect the evaluation and conclusion reached by the staff in the SER. This review was tracked by TAC M77061.

# 13 CONDUCT OF OPERATIONS

## 13.2 Training

### 13.2.1 Licensed Operator Training Program

Generic Letter 87-07, "Information Transmittal of Final Rulemaking for Revisions to Operator Licensing - 10 CFR 55 and Conforming Amendments," gives licensees the option of substituting an accredited, system-approach-to-training-based (SAT-based) training program for initial and requalification training programs previously approved by the NRC. This option is implemented upon written certification that the substitute licensed operator training program is both accredited and SAT based. Furthermore, this option allows licensees to make necessary revisions to accredited, SAT-based training programs without review and approval by the NRC.

By letter dated November 27, 1991, the applicant certified that the licensed operator training programs have been developed using a SAT-based training program, are accredited by the National Nuclear Accrediting Board, and have been implemented using a certified simulation facility. Further, the applicant committed to revise FSAR Chapter 13 to reflect the certified licensed operator training programs. The certification and commitment to revise the FSAR represent an increase in the licensed operator training program requirements, and are acceptable. The staff has reviewed Amendment 70 and confirms that the FSAR has been accordingly revised. This review was tracked by TAC M77061.

## 13.5 Plant Procedures

### 13.5.2 Operating and Maintenance Procedures

#### 13.5.2.1 Emergency Operating Procedures Generation Package

Outstanding Issue 28 was opened to track the staff's review of Watts Bar's emergency operating procedures generation package (PGP). However, the staff no longer performs such reviews but has made the emergency operating procedures development program review part of the staff's ongoing inspection program under Inspection Procedure 42001, "Emergency Operating Procedures." The inspection will be performed before issuance of an operating license. On such basis, Outstanding Issue 28 is no longer needed.

# 14 INITIAL TEST PROGRAM

While reviewing the FSAR (as revised up to Amendment 67), the staff noted two issues and asked the applicant to address them (letter, P. S. Tam to D. A. Nauman, June 6, 1991). In its response dated July 10, 1991, TVA committed to (1) reinstate the loss-of-offsite-power test for Unit 2 and (2) revise the acceptance criteria for the reactor building purge system air flow rate. The staff found these commitments acceptable in a letter dated January 13, 1992.

Subsequently, the applicant submitted Amendment 69 to the FSAR. Among other things, Amendment 69 revises the FSAR to reflect the applicant's commitment in the July 10, 1991, letter. The staff will report findings of its Amendment 69 review in a future SSER. This review was tracked by TAC M63651, M77061, M82644, and M82645.

# 15 ACCIDENT ANALYSIS

## 15.4 Radiological Consequences of Accidents

### 15.4.1 Loss-of-Coolant Accident

In Table 15.2 of the SER, the staff incorrectly identified the filter efficiency for organic iodine as 95 percent. The filter efficiency for the emergency gas treatment system should be 99 percent for organic iodine, as identified and evaluated in Section 6.5.1 of the SER. This correction is made to ensure consistency in the SER, and to identify the correct efficiency used in the staff's evaluation and the applicant's analysis (FSAR Table 15.5-6 as revised up to Amendment 63). The evaluation and conclusion reached in the SER and its supplements remain unchanged by this correction. This review was tracked by TAC M77061.

## APPENDIX A

## CHRONOLOGY OF RADIOLOGICAL REVIEW OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2, OPERATING LICENSE REVIEW

NRC Letters and Summaries

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| October 10, 1990 | Memorandum (publicly available), L. B. Marsh to P. S. Tam, transmitting report of audit performed between August 6 and 9, 1990. |
| June 6, 1991 | Letter, P. S. Tam to D. A. Nauman (TVA), forwarding interim safety evaluation of Final Safety Analysis Report (FSAR) Section 14.2. |
| October 10, 1991 | Letter, P. S. Tam to D. A. Nauman (TVA), requesting additional information on Three Mile Island (TMI) Item II.D.1, safety and relief valve testing. |
| October 11, 1991 | Summary by P. S. Tam of management meeting of September 30, 1991. |
| October 11, 1991 | Letter, P. S. Tam to D. A. Nauman (TVA), informing of site visit to review Outstanding Issue 20(a), feedwater shock valve slam. |
| October 25, 1991 | Summary by P. S. Tam of October 23, 1991, licensing status meeting. |
| November 20, 1991 | Letter, F. J. Hebdon to D. A. Nauman (TVA), acknowledging receipt of third annual report on Employee Concerns Special Program implementation. |
| November 21, 1991 | Summary by P. S. Tam of November 19, 1991, licensing status meeting. |
| November 27, 1991 | Letter, P. S. Tam to D. A. Nauman (TVA), requesting additional information on Master Fuse List Special Program. |
| December 6, 1991 | Letter, P. S. Tam to D. A. Nauman (TVA), stating that revised information submitted by TVA does not alter conclusions made in SER Supplements (SSERs) 5 and 6 concerning TMI Item III.D.1.1. |
| December 9, 1991 | Letter, P. S. Tam to D. A. Nauman (TVA), informing of resolution of technical issues in the civil/seismic areas. Results to be published in SSER 8. |

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| December 26, 1991 | Summary by P. S. Tam of December 18, 1991, licensing status meeting. |
| January 6, 1992 | Summary by P. S. Tam of December 13, 1991, management meeting. |
| January 8, 1992 | Letter, F. J. Hebdon to D. A. Nauman (TVA), transmitting copies of SSER 8. |
| January 13, 1992 | Letter, P. S. Tam to D. A. Nauman (TVA), accepting commitment to revise certain aspects of FSAR Chapter 14. |
| January 23, 1992 | Summary by P. S. Tam of January 17, 1992, licensing status meeting. |
| January 23, 1992 | Report by Thomas Cheng of April 15-19, 1991, Civil Calculation Program audit. |
| January 24, 1992 | Letter, P. S. Tam to D. A. Nauman (TVA), confirming site audit of corrective action program (CAP) on cables. |
| January 27, 1992 | Letter, P. S. Tam to D. A. Nauman (TVA), stating staff position on ductility ratio for structural steel. |
| January 30, 1992 | Summary by P. S. Tam of meeting on status of several top priority issues. |
| January 31, 1992 | Letter, P. S. Tam to D. A. Nauman (TVA), transmitting report on Civil Calculation Program audit. |
| February 5, 1992 | Letter, P. S. Tam to D. A. Nauman (TVA), accepting system-approach-to-training (SAT)-based training program. |
| February 10, 1992 | Letter, P. S. Tam to D. A. Nauman (TVA), confirming site audit to close open issues identified in Inspection Reports 50-390, 391/89-14 and 90-05. |
| February 12, 1992 | Letter, P. S. Tam to D. A. Nauman (TVA), informing of fire protection reviewer's site visit. |
| February 25, 1992 | Summary by P. S. Tam of February 19, 1992, licensing status meeting. |

TVA Letters

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| May 23, 1979 | Letter, J. E. Gilleland to NRC, forwarding response to NRC questions on containment sump and containment spray pump. |
| November 18, 1988 | Letter, S. A. White to NRC, submitting the Seismic Analysis CAP. |
| June 29, 1989 | Letter, M. J. Ray to NRC, providing additional information regarding the Seismic Analysis CAP. |

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| September 5, 1989 | Letter, M. J. Ray to NRC, providing additional information regarding the Seismic Analysis CAP. |
| July 10, 1991 | Letter, E. G. Wallace to NRC, providing information to address staff concerns on FSAR Section 14.2 regarding startup tests. |
| August 22, 1991 | Letter, J. H. Garrity to NRC, providing additional information to support the staff's review of FSAR Section 2.5, Amendments 54-63. |
| October 10, 1991 | Letter, J. H. Garrity to NRC, informing of plan to adopt the Vantage 5-H fuel design. |
| October 10, 1991 | Letter, J. H. Garrity to NRC, providing additional information on multi-mode factor of 1.2. |
| October 16, 1991 | Letter, J. H. Garrity to NRC, responding to staff request for additional information on Outstanding Issue 19(j), structural issues. |
| October 16, 1991 | Letter, J. H. Garrity to NRC, providing test report on shallow undercut concrete anchors. |
| October 16, 1991 | Letter, J. H. Garrity to NRC, providing additional information on the QA Records CAP. |
| October 23, 1991 | Letter, E. G. Wallace to NRC, requesting authorization to use American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-491. |
| October 28, 1991 | Letter, E. G. Wallace to NRC, providing schedule to implement emergency response data system. |
| October 29, 1991 | Letter, J. H. Garrity to NRC, providing additional information regarding compliance with Regulatory Guide (RG) 1.97. |
| November 4, 1991 | Letter, J. H. Garrity to NRC, addressing use of ASME Code Section XI regarding prerservice inspection programs. |
| November 8, 1991 | Letter, J. H. Garrity to NRC, transmitting Revision 4 to the Design Basis Verification Program CAP. |
| November 14, 1991 | Letter, J. H. Garrity to NRC, submitting FSAR Amendment 68. |
| November 15, 1991 | Letter, J. H. Garrity to NRC, providing additional information on implementation of Bulletin 88-01. |
| November 19, 1991 | Letter, J. H. Garrity to NRC, providing correction to letter dated October 4, 1984, on TMI Item III.D.1.1. |

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| November 21, 1991 | Letter, J. H. Garrity to NRC, transmitting report on tests done on cables removed from Watts Bar and Browns Ferry. |
| November 25, 1991 | Letter, E. G. Wallace to NRC, discussing deviations from Employee Concerns Special Program. |
| November 27, 1991 | Letter, J. H. Garrity to NRC, providing additional information on thermal evaluation criteria for structural steel. |
| November 27, 1991 | Letter, J. H. Garrity to NRC, certifying operator training program developed using SAT-based program. |
| December 2, 1991 | Letter, J. H. Garrity to NRC, informing of complete implementation of Seismic Analysis CAP. |
| December 4, 1991 | Letter, E. G. Wallace to NRC, transmitting Revision 1 of the TVA Quality Assurance (QA) Plan. |
| December 6, 1991 | Letter, J. H. Garrity to NRC, transmitting Revision 4 of the QA Records CAP and responses to NRC questions. |
| December 20, 1991 | Letter, J. H. Garrity to NRC, responding to Generic Letter 88-20, Supplement 4, on Individual Plant Evaluation of External Events (IPEEE). |
| January 15, 1991 | Letter, M. J. Burzynski to NRC, providing Revision 2 to TVA QA Plan. |
| January 14, 1992 | Letter, J. H. Garrity to NRC, providing additional information in response to Bulletin 88-05. |
| January 21, 1992 | Letter, J. H. Garrity to NRC, submitting Amendment 69 to the FSAR. |
| January 23, 1992 | Letter, J. H. Garrity to NRC, addressing schedule for submitting information on steamline break outside containment. |
| January 24, 1992 | Letter, J. H. Garrity to NRC, providing updated information on TMI Item II.F.2, instrumentation to detect inadequate core cooling. |
| January 31, 1992 | Letter, J. H. Garrity to NRC, providing response to Bulletin 90-01. |
| January 31, 1992 | Letter, J. H. Garrity to NRC, providing additional information on the Master Fuse List CAP. |
| February 2, 1992 | Letter, J. H. Garrity to NRC, providing Revision 4 to the Replacement Items Program CAP. |
| February 2, 1992 | Letter, J. H. Garrity to NRC, providing the Watts Bar Fire Protection Report. |

| | |
|---|---|
| February 12, 1992 | Letter, J. H. Garrity to NRC, responding to questions on Revision 4 of the QA Records CAP. |
| February 13, 1992 | Letter, J. H. Garrity to NRC, committing to re-perform the preoperational tests per RG 1.68, Revision 2. |
| February 25, 1992 | Letter, J. H. Garrity to NRC, requesting authorization to use an alternative to ASME Code Section III, Subsection NC/ND, Paragraph 7153. |
| February 26, 1992 | Letter, J. H. Garrity to NRC, providing Westinghouse report WCAP-12374 on Eagle-21 electronics. |
| February 26, 1992 | Letter, J. H. Garrity to NRC, providing commitment to re-analyze performance of the emergency core-cooling systems in a small-break loss-of-coolant accident. |
| April 13, 1992 | Letter, J. H. Garrity to NRC, providing copy of Norris Laboratory Report WR28-2-85-131 on containment sump screen performance. |

APPENDIX B

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Marsh, L. B., "Audit on Watts Bar Seismic Analysis CAP," October 10, 1990.

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---, Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

# APPENDIX E

## PRINCIPAL CONTRIBUTORS

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# APPENDIX G

## ERRATA TO WATTS BAR SAFETY EVALUATION REPORT

| Section | Page | Change |
|---|---|---|
| 1.1 (SER) | 1-5 | Date shown in second complete paragraph, "December 28, 1982" should read "December 28, 1981." |
| 2.5.5 (SER) | 2-42 | Top of the page, "reservoir elevation 685 ft" should read "reservoir elevation 675 ft." |
| 2.1 (SER) | 3-2 | Delete first sentence in last paragraph. Correct second sentence to read "The staff will review the revised Tables 3.2-2a and 3.2-2b...." |
| 3.6.1 (SER) | 3-14 | The reference to "FSAR 3.6.2" in the third paragraph is incorrect. It should read "FSAR 3.6A.2." |
| 7.7.4 (SER) | 7-23 | Last sentence in the third paragraph of this section currently reads "...to close the main feedwater isolation valve." It should read "...to close the main feedwater regulating valve." |
| 8.3.2.2 (SER) | 8-2 | The last complete paragraph on this page erroneously referenced a letter dated "January 15, 1984." It should read "January 17, 1984." |
| 13.4 (SSER 8) | 13-1 | "Proposed License Condition 26" should read "Proposed License Condition 25." |
| Appendix C (SER) | C-9 | First full paragraph on this page erroneously referenced "Section 10.2." It should read "Section 10.3." |
| Appendix H (SSER 4) | 1 | Item 1(b), "...and tempered 2.5 in. thick...." should read "...and tempered 1.5 in. thick...." |

# APPENDIX S

## REPORT ON AUDIT TO CLOSE ISSUES LEFT OPEN IN INSPECTION REPORTS 50-390, 391/89-14 and 90-05

AUDIT DATE: FEBRUARY 24-28, 1992

REPORTING DATE: MAY 14, 1992


Audit Team: J. R. Fair (NRC Staff, team leader)
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## AUDIT OF THE WATTS BAR CORRECTIVE ACTION PROGRAMS

### 1. BACKGROUND

In Section III of the Nuclear Performance Plan, Volume 4, the Tennessee Valley Authority (TVA) discussed its corrective action programs (CAPs) for the Watts Bar Nuclear Plant, Unit 1. TVA established the CAPs to resolve the numerous deficiencies that had been identified at Watts Bar as documented in NRC inspection reports, condition adverse to quality reports (CAQRs), nonconformance reports (NCRs), problem identification reports (PIRs), significant condition reports (SCRs), employee concerns, and internal and external reviews.

In a letter from M. J. Ray (TVA) to USNRC, dated July 18, 1989, TVA listed all of the technical issues, identified in the forgoing sources, that were to be resolved in the implementation of the CAPs.

The NRC staff and its consultants (the team) conducted two inspections of the resolution of technical issues in the CAPs related to the structural design of piping systems; cable tray/ supports; conduit/supports; equipment seismic qualification; and heating, ventilation, and air conditioning (HVAC) systems. The NRC staff documented its findings from results of these inspections in NRC Inspection Reports 50-390, 391/89-14, and 50-390, 391/90-05.

### 2. SCOPE

The scope of this audit was to review TVA's actions taken to respond to the open inspector followup items (IFIs) and unresolved items (URIs), identified in the previous CAP inspections referenced above. This included nine IFIs and two URIs. Although IFI 89-14-02 was closed in Inspection Report 50-390, 391/90-22, it was reviewed during this audit since the topic was still open under URI 88-01-02. One open item identified during the previous CAP inspections, URI 90-05-01, was not reviewed during this audit because it had been closed in Inspection Report 50-390, 391/90-24.

### 3. SUMMARY

As a result of this audit, eight IFIs and two URIs were considered closed. The item that is still considered open as a result of this audit is URI 88-01-02, which replaced IFI 89-14-02. The concern identified during this audit is that TVA has not adequately demonstrated that the cable tray covers are qualified for the dynamic loads caused by the design earthquake. TVA has considered certain "bounding cases" and applied peak spectral accelerations. However, TVA did not demonstrate that these cases were sufficiently conservative to justify the seismic response of the tray-cable-cover configuration. The cases analyzed, for example, did not consider the potential impact loads of the cables on the cover. The details of the audit team's review and evaluation of the 11 open items are discussed in Section 4. Also presented for each issue is the basis for its closure if applicable.

## 4. AUDIT DETAILS

IFI 89-14-01

Thirty-two source issues relating to cable tray and cable tray support design/ installation adequacy were identified in the TVA corrective action program (CAP) matrix provided in the TVA letter to the U.S. Nuclear Regulatory Commission dated July 18, 1989 (Reference 1). The team reviewed the actions proposed by TVA to resolve these issues during a special inspection at the Watts Bar site in September 1989, as documented in Inspection Report 50-390, 50-391/89-14 (Reference 2).

In that inspection report (Reference 2), the team found that 19 of the source issues were adequately addressed. Of the 13 remaining issues, 12 were designated as IFI 89-14-01 and Source Issue 28 was designated as IFI 89-14-02. The 12 issues were combined into one inspector followup item because the revisions to the design criteria and walkthrough procedure, needed to address these issues, were not completed at that time. The disposition of each of the source issues under IFI 89-14-01 follows.

The source issues to be addressed under this inspector followup item can be grouped into two categories; those associated with deficiencies or omissions in the design criteria and those associated with the adequacy of the as-built installations. A TVA closure document for this IFI acceptably summarizes the source issues and their associated source documents grouped into two categories as indicated in Table 1.

In the subsequent subsections related to IFI 89-14-01, the 12 source issues listed in Table 1 are discussed in detail. Since Table 1 presents a brief description of the source issues, the following discussion focuses on the review and evaluation pertinent to each of the 12 source issues.

TABLE 1: ISSUES RELATED TO IFI 89-14-01

DESIGN CRITERIA ISSUES

| Source Issue No. | Source Document | Description |
|---|---|---|
| 2 | CAQR WBP880040<br>NRC Violation<br>390,391/88-01-02 | ZNB and ZNK (i.e., fittings and offset-type fittings, respectively) have not been qualified for various field configurations. |
| 4 | CAQR WBP880418<br>NRC Violation<br>390,391/88-01-02 | No documentation exists to qualify cable tie wraps for horizontal cable trays mounted on their sides. No documentation exists to qualify installation of cable tray covers during a seismic event. |
| 5 | CAQR WBP870818 | Cable trays have not been evaluated for differential movement between buildings. |
| 7 | DR-89 | No documentation can be found confirming that designers used accurate weights for cable tray support design to include cable, covers, etc. |
| 25 | DR-464 (5B through 5E) | Discrepancies have been found with cable tray connectors and fittings. |

OVERINSPECTION ISSUES

| Source Issue No. | Source Document | Description |
|---|---|---|
| 1 | CAQR WBP870528<br>CATD 11103-WBN-08<br>IN-85-865-002 | Adequate documentation was not maintained for the closure of NCR 5737 R1. Not all cable tray supports were walked down to compare as-built configurations with issued drawings during response to NRC 5737 R1. |
| 6 | SCR SQNCEB8622 | Cable tray support design issues identified at SQN. Verify the potential generic condition evaluation performed for WBN. |
| 10 | DR-237 (5B) | Grout is chipped or cracked, or both, under baseplates. |
| 18 | DR-437 (5A,B) | G-32 anchor bolt spacing and minimum edge distance requirements are violated. |
| 21 | DR-447 | Support is installed in incorrect location and has dimensional discrepancy. |
| 26 | DR-472 (5A,B,D,E) | Cable tray support is not installed per approved details. |
| 27 | DR-473 (5A,5B) | Baseplate is different size than specified on drawing. One baseplate has chipped grout in one area. |

DESIGN CRITERIA ISSUES

Issue No. 2

Review

TVA's action to resolve this issue includes a complete walkthrough of all cable trays. One aspect of the walkthrough was to evaluate field configurations involving ZNB and ZNK fittings and associated hardware. ZNB fittings are adjustable riser fittings allowing hinge action about an axis lying on the plane of the cable tray. ZNK fittings are horizontal fittings with the hinge axis perpendicular to the cable tray plane. The team reviewed pertinent qualification procedures for both of the fittings during this audit.

Evaluation

Singleton Materials Engineering Laboratory tested ZNK hinged-cable tray connectors (Reference 3). These tests include direct tension, bending, and in-plane and out-of-plane shear capacity evaluation. According to Reference 3, the typical failure observed in these tests involved pulling and subsequent straightening out of the fingers of the connector around the pin. This behavior seems reasonable in view of the loading conditions considered in the qualification procedure. During the audit, it was unclear by looking at copies of the photographs taken from these tests whether the test configuration was identical to that observed in the field configuration at the plant. TVA provided a set of the original photographs from which the team concluded that both configurations were identical. The team reviewed details regarding the implementation of the test results into the screening criteria used to evaluate the plant cable tray configurations; these were reviewed and found acceptable. The team also reviewed TVA's evaluation for the ZNB fittings. The latter are also qualified by tests (Reference 4). Both straight as well as 45°-angle connectors were tested to determine failure loads that are consistent with the loading conditions required by the design criteria.

The team found that the cable tray qualification program being implemented by TVA was acceptable in terms of resolving the issues raised with respect to ZNB and ZNK fittings. The finding that both fittings were adequate was based on qualification by tests, which is an acceptable approach per general design criteria. On this basis, the team concludes that this source issue has been adequately addressed.

Source Issue No. 4

Review

This issue is not being reviewed under IFI 89-14-01 because the deficiencies, concerns, and questions expressed by Source Issue No. 4 are addressed in IFI 89-14-02 (cable tray issues). Accordingly, this issue is reviewed in the corresponding section of IFI 89-14-02 of this audit report.

Source Issue No. 5

Review

The basic concern expressed by this source issue is that cable trays spanning from one building to another were not evaluated for differential seismic movements between buildings. This concern was also expressed in Reference 5 for other subsystems, e.g., Category I piping and HVAC ducts. However, in the context of IFI 89-14-01, only the case of cable trays is discussed. During the audit, the team reviewed actions taken by TVA to address this issue. According to TVA, the issue of accommodating building differential movements during the design seismic event, was identified through TVA's efforts related to the Integrated Interaction Program (IIP). Specifically, as part of the shakespace program, all crossings between buildings were identified. The corresponding cable tray configurations were identified using the walkthrough procedure WD-019. Consequently, TVA evaluated and screened the cable tray configurations subjected to differential seismic movements between buildings using WD-DC-20-32 procedures and criteria. The outcome of the above efforts was that about 35 cases involving crossings of cable trays between buildings were identified as requiring field modifications in order to allow for differential building displacement. The team also reviewed some examples of proposed shakespace modifications as described in DCN M-16295-A.

Evaluation

The overall approach followed by TVA to address this issue is acceptable. Specifically, the screening procedures are expected to identify the cable trays subjected to the requirement of accommodating the differential seismic displacement between buildings. The other aspect of this issue is related to the efficiency of the proposed modifications in performing their required function. From a review of some of the proposed changes, it appears that additional supports will be placed near the gaps so that possible local amplifications would be avoided (e.g., cantilever action of portions of the cable trays between the gap and the existing nearest support). Such amplifications could alter the 2-inch gap which was determined from the seismic evaluation of the plant. The team finds TVA's approach to provide additional supports in the neighborhood of the proposed gaps appropriate.

In conclusion, the proposed procedures to address the cable tray differential seismic movement between buildings have adequately addressed this source issue.

Source Issue No. 7

Review

During this audit, the team reviewed the design criteria document WB-DC-20-21.1 (Reference 6) to confirm that the loads indicated by this issue are prescribed in the general criteria. The team concluded that Section 4.2.1 of the design criteria related to dead loads includes combined weights of cables, covers, and protec-tive coatings. In order to keep track of the loads on the cable trays at the field and confirm that they do not violate the design loads, TVA employs a computerized system. During the audit, the process of confirming that cable trays are subjected to proper design weights was discussed in detail. The key element of this process is based on a design weight of 30 psf. Accordingly, a Computerized Cable Routing System (CCRS) is documenting the cable fill, and any overfills that would violate the design weight are transmitted through the quality information request process to civil engineers for evaluation.

Evaluation

According to TVA, the design weight was determined by taking into account pertinent weight contributors that were outlined in the cable tray design criteria. This includes the weights of the cables, trays, fittings, splices, covers, and protective coatings. TVA implemented a system to monitor conformance with the design weight. This system is expected to identify potential overweight situations. The actions taken by TVA have adequately addressed this source issue.

Source Issue No. 25

Review

TVA based its action to resolve this issue on a qualification program for all tray hardware, including connectors and fittings for cable trays located in Category I buildings. This program is being implemented using the cable tray design criteria (Reference 6, Appendix A). TVA's strategy for addressing this issue is to review as-built conditions of the cable tray hardware during the walkthrough, and proceed with an evaluation based on the foregoing criteria. If criteria are not met, modifications will be made.

During the audit, it was indicated that EQE has performed a walkthrough for this purpose. As a result, about 350 modifications are expected to be made. EBASCO is expected to evaluate EQE's proposed modifications and issue appropriate DCNs.

Evaluation

The proposed approach is expected to resolve the concerns regarding inconsistencies with cable tray connectors and fittings; therefore, it is concluded that this source issue has been adequately addressed.

OVERINSPECTION ISSUES

Source Issue No. 1

Review

During the audit, pertinent details that lead to this issue were discussed. TVA indicated that it started with about 5000 supports; 3000 of which were reinspected through walkdowns as part of the corrective action related to the closure of NCR 5737 R1. This activity took place around 1984. At that time, the remaining 2000 supports were not ready for walkdowns to evaluate their as-built condition. There appear to be two main aspects raised by Source Issue No. 1. One questions the adequacy of the documentation employed for the first batch of 3000 supports (closure of NCR 5737 R1) and the other raises concerns about the lack of reinspection of the remaining 2000 cable tray supports.

Evaluation

TVA's action taken to resolve this issue was twofold. First, in order to address the question about the adequacy of the documentation of the 3000 cable tray supports previously reinspected, an overinspection walkthrough was performed on a sampling basis. Specifically, 58 of these supports were reconsidered. The results of this activity are being evaluated within the framework of the corrective action program. Second, in response to the concern about the remaining 2000 supports, TVA performed a walkthrough using walkthrough procedure TI-2004. Again, the results of the walkthrough are being evaluated within the framework of the corrective action program. The procedures used by TVA to address Source Issue No. 1 are considered acceptable and are expected to answer the concerns raised.

Source Issue No. 6

Review

During the audit, the actions taken by TVA to resolve the question of possible applicability of issues related to the Sequoyah plant were reviewed. A total of six issues were identified at Sequoyah (SCR No. SQNCEB8622) concerning such items as loading conditions, overspan, and overloaded cases. As a result of TVA's evaluation of these issues, it was concluded that such conditions were generally applicable to Watts Bar. The specific actions taken to resolve these issues were reviewed during the audit.

Evaluation

The specific issues considered by TVA for resolving the source of this concern include (1) loading conditions, (2) applicability of response spectra, (3) embedded and anchor plates, (4) overloaded trays, (5) cable tray spans, and (6) calculations supporting field modifications. The approach taken by TVA to resolve the potential generic condition of the issues identified at Sequoyah is comprehensive and is expected to resolve the source of this concern. Thus, this source issue has been adequately addressed.

Source Issue No. 10

Review

This issue is concerned with deficiencies in grouted anchor plates. Specifically, it was found that the grout was chipped or cracked or both at various locations. According to TVA, this discrepancy was corrected by repairing the damaged grout. Furthermore, possible generic implications from this issue were addressed by considering the chipped/cracked grout under baseplates as an attribute for cable tray support walkthroughs.

Evaluation

The action taken by TVA to address the concern for the specific case of damaged grout under a baseplate is acceptable. In addition, TVA's action to subsequently consider this issue as an attribute for cable tray support walkthroughs has adequately addressed this source issue.

Source Issue No. 18

Review

This issue concerns discrepancies in minimum anchor bolt spacing requirements. In order to resolve the specific cases where such discrepancies were identified, TVA performed stress calculations (Reference 7) using the corresponding "as-built" configurations. The team reviewed these calculations during the audit.

Evaluation

The action taken by TVA to resolve this issue was to check structural adequacy of the as-built condition. On the basis of TVA's calculations, the team concluded that the existing configurations satisfy stress allowables per design criteria without a need for modifications. Thus, this source issue has been adequately addressed.

Source Issue No. 21

Review

This issue concerns a discrepancy in location and configuration of a cable tray support. In response to this issue, TVA revised the corresponding drawing to reflect the correct location (Drawing 48W1296-1).

Evaluation

During the audit, the team reviewed the revisions made by TVA and found them acceptable. TVA's action taken to resolve this source issue is adequate.

Source Issue No. 26

Review

The discrepancies identified by this issue are associated with cable tray supports. Specifically, concerns were raised with regard to violation of G-32 requirements (Reference 8); discrepancy between "as-built" configuration and design drawing; and finally, undocumented attachments to the cable tray support. During the audit, the team reviewed TVA's evaluation of the identified discrepancies. The basis of the evaluation was to perform stress checks to demonstrate the adequacy of the "as-built" condition.

Evaluation

The result of TVA's evaluation is that the existing configuration satisfies the design criteria. TVA's action to resolve this issue was based on a stress evaluation of the "as-built" conditions. This evaluation confirms that cable tray support criteria for Watts Bar are met. On this basis, the team concludes that this source issue has been adequately addressed.

Source Issue No. 27

Review

This source issue concerns differences between anchor plate "as-built" condition and corresponding design drawings. In addition, concerns about chipped grout were also expressed (see also Source Issue No. 10). In order to resolve this issue, TVA issued DCN C-2410-A to revise the corresponding drawing to reflect the actual configuration (Drawing 48W1298-8). The baseplate was installed as 20"x 20", while the drawing showed 18" x 18". During the audit it was shown that Drawing 48W1298-8 was revised to show the actual size of the baseplate (20" x 20"). The grout under MK6P was repaired per MR No. A-606089.

Evaluation

TVA's action taken to resolve this issue is acceptable. TVA had revised design drawings to reflect the actual support condition and the damaged grout was repaired. Thus, this source issue has been adequately addressed.

Overall Conclusion for IFI 89-14-01

On the basis of the above discussions, the 12 source issues reviewed under IFI 89-14-01 have been adequately addressed. Thus, IFI 89-14-01 is considered closed. As indicated previously, Source Issue No. 4 is now being reviewed under IFI 89-14-02.

References


1. TVA Letter to USNRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Corrective Action Program (CAP) Plan Matrices," dated July 18, 1989.

2. USNRC Inspection Reports 50-390, 391/89-14, Watts Bar Nuclear Plant, Units 1 and 2, September 1989.

3. "Sequoyah Nuclear Plant Load Capacity Testing of Cable Tray Adjustable Horizontal Connector (TVA ID ZNK)," SME-STR-88-004; March 18, 1988.

4. "Cable Tray Adjustable Riser Connector Assembly Used at Watts Bar and Matching 18-inch Tray," Singleton Materials Engineering Laboratory, CVB-85-3, August 7, 1985.

5. Source Document: CAQR WBP 870818, June 1988.

6. "Category I Cable Tray Supports - General Design Criteria," TVA WB-DC-20-21.1, August 1976.

7. "Cable Tray Support 0-CTSP-292-0602," WCG-AB-1298-0602, Ebasco Services, Inc., August 1991.

8. "TVA General Construction Specification G32, Bolt Anchors Set in Hardened Concrete," Rev. 12.

IFI 89-14-02 (URI 88-01-02)

Description of Issue(s)

The CAP matrix has identified a total of 32 issues associated with the design of cable trays as well as cable tray supports for the Watts Bar Nuclear Plant. In particular, Source Issue No. 28 was concerned with a cable tray cover that was attached to the corresponding cable tray with a wire. During the NRC's inspection which was conducted from September 12 through 18, 1989, this source issue was not reviewed because TVA, at that time, was in a process of weighing different alternatives for resolving this issue. Since the resolution of the cover issue had not been determined by TVA at the September 1989 inspection, it was decided to undertake the final disposition of this issue during a followup inspection. TVA's letter to NRC dated August 30, 1990 (Reference 1) indicates that TVA was ready for NRC closure regarding this issue.

It should be noted that Source Issue No. 4 under IFI 89-14-01 also deals with cable tray covers. Since the concerns expressed by both issues are essentially the same, Source Issue No. 4 of IFI 89-14-01 is considered closed with the understanding it is treated under IFI 89-14-02. Although IFI 89-14-02 was closed in Inspection Reports 50-390 and 391/90-22, it was reviewed during this audit since the topic was still open under URI 88-01-02.

Review

During this audit, the proposed approach was discussed and pertinent documentation was evaluated. A significant effort was made to deal with the details of the cover issue and evaluate the adequacy of the proposed approach.

As part of TVA's response to resolve this issue, construction details for installation of cable tray covers were issued according to Design Change Notice (DCN) M-10472A (Reference 2). This DCN calls for use of cover straps to address the concern regarding the cover's adequacy during a seismic event. A set of configurations is proposed to treat the tray cover problem for different cable tray applications, for example, channel connectors, strap connectors, and raised cover connectors. A key element of TVA's position for addressing this issue is that tie-wraps have been determined to be inadequate to support the cables. Accordingly, TVA's interpretation is that cable tray covers are employed for structural reasons related to position retention of the cables in the trays. According to TVA, there are additional nonstructural reasons for using cable tray covers in the plant. These include such requirements as separation, fire protection, and protection from physical damage. (WB-DC-30-22, "Electrical Raceways," describes five general cases on which cable tray covers should be provided for cable trays and risers.)

The modifications proposed in Reference 1 were supplemented with supporting calculations (Reference 3). The latter include horizontal trays, risers, and side-mounted trays which are primarily found in the annulus area of the reactor building. In these calculations, the various details for attaching the covers into the cable trays were considered and their structural adequacy was evaluated.

Evaluation

Responding to questions regarding the capability of the covers to withstand the expected loading due to the postulated design seismic event, TVA indicated that peak spectral accelerations were used during the qualification process. In addition, TVA pointed out that the damping is expected to be higher than that assumed in the analysis. Although such arguments seem reasonable, a conclusion could not be reached regarding the amount of conservatism that such arguments introduce. Specifically, taking such conservatism into account, it could not be demonstrated, in a quantitative manner, that the expected dynamic behavior of the tray-cable-cover system during a seismic event would be sufficiently estimated. The cases of interest primarily involved side-mounted cable trays that are most often found in the annulus area. To further assess the validity of the equivalent static analysis employed in the qualification procedure, a two-degree-of-freedom simplified model was considered during the audit to simulate the dynamic behavior of the tray-cable-cover system. The team concluded that several questions can be reasonably addressed by using such a simplified model. However, preliminary results presented during the audit were based on gross assumptions and were not considered adequate. One of the key questions is related to postulated impact loads acting on the cable tray covers during a seismic event. TVA indicated that, in view of this, it will consider a verification plan with the objective to verify the adequacy of the existing design of cable tray covers.

As a result of the audit, the team concluded that additional justification is needed to fully resolve the cable tray cover issue. Specifically, TVA is considering a verification study to demonstrate the adequacy of the seismic design of cable tray covers and associated connector hardware. At this stage, a complete evaluation cannot be made until the results of the verification study become available for review. On the basis of the review conducted thus far, it appears that TVA's efforts to address this issue are reasonable and are expected to produce acceptable results for final resolution of the issue. As indicated earlier, IFI 89-14-02 was closed in Inspection Reports 50-390, 391/90-22. Thus, the resolution of this issue will be tracked under URI 88-01-02.

References

1. TVA Memo to G.A. Walton, NRC Senior Resident Inspector: "Watts Bar Nuclear Plant (WBN) - Nuclear Regulatory Commission -

Inspector Follow-up Item (IFI) 89-14-02," from P.L. Pace, TVA, August 30, 1990.

2. Design Change Notice, Division of Nuclear Engineering, "Category I Cable Tray Covers," Watts Bar Nuclear Plant - Unit 1, DCN No. M-10472A, June 13, 1990.

3. Calculation No. WB-CT-01A: "Cable Tray Cover Connectors Qualification," Sargent & Lundy Engineers, May 1990.

URI 89-14-04, IFI 89-14-06, IFI 89-14-08, and IFI 89-14-09

Description of Issues

Deficiencies in the design criteria for conduits and conduit supports were identified in SCR WBN CEB 8675 R1, WBRD-50-390/86-14, and DRs 313, 315, 316, 326, 469, and 500. It was stated that typical designs did not envelope worst-case design parameters and that some design configurations were not addressed by the criteria. These deficiencies were designated Source Issue No. 4 in the TVA conduit and conduit support CAP matrix (Reference 1). Discrepancies between the design criteria and FSAR commitments were also identified in CAQR WBF 870087 and CATD 22403-WBN-01. These were designated Source Issue No. 9 in the CAP matrix.

During the NRC September 1989 inspection (Reference 2), the team reviewed TVA's actions to resolve these issues. The actions, although responsive, did not resolve all aspects of the source issues. Several of the remaining or open concerns were associated with the use of the TVA load span table method of qualification analysis and have now been resolved with one TVA action. These are grouped together in this review and evaluation section.

The concerns associated with the use of the load span table method of analysis are:

- URI 89-14-04, the design considerations for conduit branch lines and bends were not clearly addressed.
- IFI 89-14-06, design criteria documentation for the consideration of concentrated masses in the load span table was incomplete.
- IFI 89-14-08, adequate documentation for the basis for the modal factors used to develop values in the load span table was not provided.
- IFI 89-14-09, the effect of rigid modes on the load span tables was not addressed.

Review

To resolve these concerns, TVA deleted from the design criteria the load span table method as an option to qualify conduit and conduit supports. TVA further advised that the walkthrough procedures used to review existing installations provide guidance to assess spans with branches and bends and defines acceptable limits for concentrated masses. Lastly, TVA advised that applicable conduit/conduit support typical drawings have been reviewed and validated to the latest design criteria requirements, including the treatment of branches, bends, and concentrated masses.

During this audit, the team reviewed the latest version of the design criteria (Reference 3) and the walkthrough procedures (References 4 and 5). Clearly, all mention of the load span table qualifica-tion option had been deleted from the design criteria. Reference to the version of the design criteria document available during the previous audit indicated that the load span table option was the qualification option of choice at that time and its deletion from the criteria represented a significant change in analysis philosophy. The pertinent source documents (References 6 and 7) and TVA calculations referenced in those documents (References 8, 9, & 10) were also reviewed to assure full appreciation of the issues and to substantiate that these documents were directly associated with the load span table method.

Evaluation

The revised design criteria allow only the use of the equivalent static analysis method and the response spectrum method as options to qualify conduit systems. In both of these methods, a geometrically correct model of the system, including explicit modeling of bends, branches, and concentrated masses, is used in the qualification calculations. Properly applied, the revised design criteria adequately address the concerns regarding the consideration of bends, branches, and concentrated masses. Further, the review of the walkthrough procedures substantiated that they did address and provide guidance for the evaluation of systems with branches, bends and concentrated masses. Regarding the modal factors and rigid modes, these were aspects of the development of the load span tables, and with the deletion of the tables these concerns are no longer of consequence. On the basis of these observations URI 89-14-04, IFI 89-14-06, IFI 89-14-08 and IFI 89-14-09 are considered closed.

References

1. TVA letter to USNRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Corrective Action Program (CAP) Plan Matrices," July 18, 1989.

2. USNRC Inspection Reports 50-390, 391/89-14, Watts Bar Nuclear Plant, Units 1 and 2, September 1989.

References - cont'd


3. "Seismically Qualifying Conduit Supports, General Design Criteria," WB-DC-40-31.10, Rev. 7, February 21, 1992, RIM No. T29 920221 898.

4. "Engineering Walkthrough and Evaluation of Plant Conduit and Conduit Supports," TI 2006, Rev. 0, September 28, 1990.

5. "Engineering Walkthrough Procedure for the Conduit and Conduit Support Critical Case Evaluation," WP-51, Rev. 0, April 3, 1989, RIM No. B26 89 0403 011.

6. "Significant Condition Report," SCRWBNCEB8675R1, November 28, 1986, RIM No. B26 870213 009.

7. "Deficiency Reports" DR 313-5, Rev. 1, December 15, 1988.

8. "Conduit Support Loads When Elbows, Tees, and Bends Exist in Any Span," TVA Calculation No. CEB MA2 093, Rev. 0, February 6, 1987, RIM No. B41 870206 004.

9. "Conduit Support Loads for Concentrated Load Applied at Any Location," TVA Calculation No. CEB MA2 012, Rev. 0, February 6, 1987, RIM No. B41 870206 005.

10. "Parametric Study of Spans," TVA Calculation No. CEB-MA2-095, Rev. 0, February 6, 1987, RIM No. 870206 002


URI 89-14-05

Description of Issue

In source document SCR WBN CEB 8675 R1, a question was raised about the adequacy of the qualification of conduit for thermal conditions. This question was one aspect of Source Issue No. 4 in the TVA conduit and conduit support CAP matrix (Reference 1). During the NRC September 1989 inspection (Reference 2), the team reviewed TVA's revisions to the design criteria to resolve this concern; the revisions were not considered sufficient. In particular, the criteria did not provide a procedure to evaluate thermal loadings on long straight runs of conduit. This concern was designated URI 89-14-05.

Review

Revisions to the design criteria document that provides guidance to address thermal loadings and specific calculations to support that guidance were prepared by TVA to resolve this concern. A review of the current revision of the design criteria for conduit systems (Reference 3) and the supporting TVA calculation (Reference 4) was performed during this audit.

Evaluation

The criteria were found to provide explicit guidance regarding conduit runs and thermal loadings in the form of a table of allowable straight lengths for which no review is required, and by specifying the load combinations and allowable stresses to be used to evaluate outlier cases. The calculation provided an analysis of straight conduits subjected to thermal loadings and adequately demonstrated that the guidelines for allowable straight lengths specified in the criteria resulted in conduit stresses that were well within allowable limits for accident cases and that normal thermal load effects were negligible.

The revisions to the design criteria document are considered to address, in an acceptable manner, the qualification of straight conduit runs subjected to thermal loadings. Applicable criteria, their limits, and the action to be taken if they are exceeded, are presented. Regarding the last, the action to be taken if the criteria are exceeded was considered to be deficient because it did not explicitly state which seismic load case or stress allowance was to be used in the evaluations. This deficiency was discussed and TVA agreed to revise the design criteria to clarify both the seismic load case and stress allowable to be used under such circumstances. The wording of a proposed revision to Section 4.9 of the design criteria (WB-DC-40-31.10) to address this deficiency was provided by TVA in NRC Audit Review Sheet RIM No. T30 920227 826, dated February 27, 1992, and was considered acceptable.

URI 89-14-05 is considered closed based on the revised design criteria and TVA's commitment to further revise the criteria in accordance with the planned action writeup in the audit review sheet referenced above.

References

1. TVA letter to USNRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Corrective Action Program (CAP) Plan Matrices," dated July 18, 1989.

2. USNRC Inspection Reports 50-390, 391/89-14, Watts Bar Nuclear Plant, Units 1 and 2, September 1989.

3. "Seismically Qualifying Conduit Supports, General Design Criteria," WB-DC-40-31.10, Rev. 7, February 21, 1992, RIM No. T29 920221 898.

4. "Straight Conduits Subject to Thermal Loads," WCG-WB-CS-2S, Rev. 1, June 14, 1991.

IFI 89-14-07

Description of Issue

Inconsistencies between the design criteria for conduit and conduit supports and the Watts Bar FSAR were identified to exist in CAQR WBF 870087, CATD 22403-WBN-01, and DRs 313 and 315. These discrepancies were designated Source Issue No. 9 in the TVA conduit and conduit support CAP matrix (Reference 1). During the NRC September 1989 inspection (Reference 2), the TVA actions to resolve these inconsistencies were reviewed and found acceptable with one reservation. At the time of the inspection, the damping for the operating-basis earthquake (OBE) and safe-shutdown earthquake (SSE) were both 7% in the proposed revision to the FSAR while they were 4% and 7%, respectively, in the design criteria document. TVA advised that the design criteria would be revised to be consistent with the FSAR when NRC approved the proposed damping. Since the inconsistency still existed, the concern remained open and was designated IFI 89-14-07.

Review

To resolve this concern, TVA revised the FSAR to specify 4% damping for the OBE and 7% damping for the SSE, values consistent with those in the design criteria.

During the audit, the team reviewed the pertinent sections of the latest amendment of the FSAR (Reference 4) and the current revision of the design criteria (Reference 3).

Evaluation

The damping values for conduit were specified as 4% for the OBE and 7% for the SSE in both the latest amendment of the FSAR and the design criteria. TVA's action has thus resolved the inconsistency that was the source of this concern. Regarding the specified damping values, they were assessed by the staff during its review of Amendment No. 64 to the FSAR (Reference 5) and were considered acceptable. On the basis of these observations, IFI 89-14-07 is considered closed.

References

1. TVA letter to USNRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Correction Action Program (CAP) Plan Matrices," July 18, 1989.

2. USNRC Inspection Reports 50-390, 391/89-14, Watts Bar Nuclear Plant, Units 1 and 2, September 1989.

3. "Seismically Qualifying Conduit Supports, General Design Criteria," WB-DC-40-31.10, Rev. 7, dated February 21, 1992, RIM No. T29 920221 898.

References - cont'd

4. Watts Bar FSAR, Section 3.10.3.3 and Table 3.7-2, Amendment 64.

5. USNRC Safety Evaluation Report, Watts Bar Nuclear Plant Units 1 and 2, Docket Nos. 50-390 & 50-391, NUREG-0847, Supplement No. 8, January 1992.

IFI 89-14-10

Description of Issue(s)

Source document CAQR WBP 870818 identified the deficiency that rigid conduit spanning between buildings was not evaluated for the differential movements that may occur between the buildings during a seismic event. This deficiency was designated Source Issue No. 12 in the conduit and conduit support CAP matrix (Reference 1). TVA's action plan to resolve this issue was reviewed during the NRC September 1989 inspection (Reference 2) and found to be deficient in two aspects. Firstly, the need to more clearly address the manner in which rigid conduit will be evaluated for the relative displacement between seismic interfaces was designated IFI 89-14-10, and is addressed in this section. Secondly, the basis for the design adequacy of flexible conduits spanning the interfaces was designated IFI 89-14-11, and is the subject of the next section.

Review

TVA's actions to resolve this concern was to revise the design criteria to clearly address the manner in which rigid conduit will be evaluated for differential seismic movements, and to implement the integrated interaction program screening review to, in part, provide criteria to assess crossings and to identify by walkthrough all crossings and assess them per the screening criteria.

The design criteria (Reference 3), the integrated interaction program screening and acceptance criteria document (Reference 4), the shakespace closure calculation (Reference 5), and the flexibility charts calculation (Reference 6) were reviewed during this audit.

Differential building movements are addressed in Section 4.7 and Appendix A3 of the design criteria. While Section 4.7 states that the design action to address crossings is to decouple the conduit segments using a flexible conduit at the crossing, Appendix A3 requires that seismic anchor motions shall be evaluated in a manner that produces worst-case response when rigid crossings occur. Discussions indicated that although the flexible conduit approach is the norm, the option to use rigid conduit crossings was still desired.

The integrated interaction program included a shakespace interaction screening review through which crossings between buildings for all commodities were identified and assessed. The document included a listing of building movements to be accommodated and criteria, in the form of flexibility charts, to assess the adequacy of rigid crossings (i.e., crossings made without the use of flexible components).

The shakespace closure report documents the results of the integrated interaction program, providing summaries of the walkthroughs, calculations to evaluate outlier configurations, and descriptions of necessary modifications. A review of the data for conduit crossings indicated that the majority of these were made with flexible conduit, and the few that were rigid crossings, could not be shown to meet design criteria. For these rigid crossings, modifications will be necessary, most likely incorporating flexible conduit elements.

The flexibility chart calculations were performed to develop charts of the lengths of different commodities required to accommodate differential anchor movements. The calculations involved the application of simple engineering formulae and the appropriate stress allowables. The charts thus developed were used to screen the commodity crossings.

Evaluation

As noted above, it is stated in the design criteria that rigid conduit, supported between seismic interfaces, should be decoupled by use of flexible conduit. Further, in the design of the crossing, all applicable building movements and settlements must be considered. This design procedure is acceptable.

The criterion to qualify rigid conduit crossings, included in Appendix A3 of the design criteria, on the other hand, was considered to be deficient. It did not explicitly define either the load cases or stress allowables to be used in the evaluation process. The observation of deficiency was conveyed to TVA. TVA responded by preparing the NRC Audit Review Sheet RIM No. T30 920227825, dated February 27, 1992, which presented, under the planned action writeup, a proposed revision to the design criteria to address the perceived deficiency. The proposed revision was considered acceptable.

The TVA efforts to identify and assess all conduit crossings through the integrated interaction programs is commendable. The program appears to be both extensive and comprehensive.

IFI 89-14-10 is considered closed based on the revised design criteria and TVA's commitment to further revise the criteria in accordance with the planned action writeup in the audit review sheet referenced above.

References


1. TVA letter to USNRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Corrective Action Program (CAP) Plan Matrices," July 18, 1989.

2. USNRC Inspection Report 50-390, 391/89-14, Watts Bar Nuclear Plant, Units 1 and 2, September 1989.

3. "Seismically Qualifying Conduit Supports, General Design Criteria," WB-DC-40-31.10, Rev. 7, February 21, 1992, RIM No. T29 920221 898.

4. "Integrated Interaction Program Screening and Acceptance Criteria," WB-DC-20-32, Rev. 0, April 12, 1991, RIM No. B26 91 0412 112.

5. "Shakespace Closure Calculation," 50052-C4-108, Rev. 0, July 29, 1991, RIM No. B18 910808 035.

6. "Flexibility Charts Calculations," 50052 C0 007, Rev. 0, February 28, 1991, RIM No. B26 910412 154.


IFI 89-14-11

Description of Issue

As noted in the previous section, TVA's action plan to resolve Source Issue No. 12 of the conduit and conduit support CAP matrix (Reference 1) was reviewed during the NRC September inspection (Reference 2) and was found deficient in two aspects. The first deficiency was designated IFI 89-14-10 and was addressed in the preceding section. The second deficiency, the lack of a basis for the design adequacy of flexible conduits spanning interfaces, was designated IFI 89-14-11, and is the subject of this section.

Review

TVA's action to resolve this concern was to make reference to the Integrated Interaction Program to substantiate the building differential movement criteria in the design criteria, and the installation specification for electrical systems, to substantiate that field installations of conduit comply with the design criteria.

In the audit, the pertinent sections of the design criteria (Reference 3), the integrated interaction program screening and acceptance criteria document (Reference 4), and the installation specification (Reference 5) were reviewed. Section 4.7 of the design criteria states that seismic differential movements, design-basis accident (DBA) and thermal movements for the steel containment vessel, and building differential settlements must be considered in the design of flexible conduits. Section 3.2.6.3 of the installation instructions specifies detailed requirements for the length of the flexible conduit spanning interspaces. The requirement provides a minimum of 1 inch of extra conduit length for each installation. Table 3-7 of the Integrated Interaction Program document summarizes the maximum building displacements to be expected between buildings at any elevation.

Evaluation

As noted above, the design criteria require that differential movements between buildings be accommodated with flexible conduits, but it does not provide any specific requirements for that design. Instead, the installation instructions provide a 1-inch extra length for the installed component. Reference to Table 3-7 of the Integrated Interaction Program document indicates that the minimum 1-inch-free-length allowance is greater, with an ample margin, than any expected displacement.

This design approach for flexible conduits is simple and direct. It essentially requires a design that can accommodate worst case movements to be enforced for every flexible conduit spanning between buildings. The approach is acceptable and IFI 89-14-11 is considered closed.

References


1. TVA letter to USNRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Correction Action Program (CAP) Plan Matrices," July 18, 1989.

2. USNRC Inspection Report Nos. 50-390, 391/89-14, Watts Bar Nuclear Plant, Units 1 and 2, September 1989.

3. "Seismically Qualifying Conduit Supports, General Design Criteria," WB-DC-40-31.10, Rev. 7, February 21, 1992, RIM No. T29 920221 898.

4. "Integrated Interaction Program Screening and Acceptance Criteria," WB-DC-20-32, Rev. 0, April 12, 1991, RIM No. B2691 0412 112.

5. "General Engineering Specification G-40," Rev. 12, July 31, 1991, RIM No. B43 91 0731 004.


IFI 90-05-02

Description of Issue

In Section 3.10 of the Watts Bar Safety Evaluation Report (NUREG-0847), Supplement No. 3 (Reference 1), the NRC identified three generic and five specific concerns regarding the seismic and dynamic qualification of safety-related electrical and mechanical equipment. In NRC Inspection Report 50-390, 391/90-05 (Reference 2), two of the generic issues and five specific issues were closed. The remaining issue, IFI 90-05-02, relates to the seismic qualification testing of all safety-related equipment supplied by Westinghouse.

The generic concern in IFI 90-05-02 is that seismic testing of equipment using single-axis excitation and single-frequency testing may not be acceptable without adequate justification. In particular, TVA was asked to provide verification of the following items:

- The effect of directional coupling should be considered if applicable.
- Where applicable, verification should be provided that acceleration at each device location is less than 0.95g because relay chatter at higher acceleration levels is expected.
- The test response spectrum (TRS) envelopes the required response spectrum (RRS) for all directions.

Review

During this audit, TVA provided several documents which describe the approach and criteria that will be followed to address the concerns of IFI 90-05-02. TVA's planned action is described in a TVA internal document entitled "NRC Audit, Equipment Seismic Qualification, 24-28 February 1992" (Reference 3). The attachment to this document contains the Watts Barr "Work Plan" and "Draft Guidelines for the Evaluation of Directional Coupling and Multi-Mode Effects for Equipment Tested to Single-Axis and/or Single-Frequency." The Work Plan presents a flow chart of the major steps to be taken to address the concerns of IFI 90-05-02. The guidelines provide a more detailed description of how to justify the single-axis or single-frequency testing of equipment, or both.

Also reviewed during this audit was TVA Design Criteria No. WB-DC-40-31.2 (Reference 4) as it relates to seismic testing. This Watts Barr Design Criteria specifies requirements for the seismic qualification of Category I fluid system components and electrical or mechanical equipment.

Evaluation

TVA's action plan, as defined in Reference 3, addresses all three subissues listed above. The major steps of the Work Plan are: (1) obtain Westinghouse supplied equipment list with reference to qualification documents, (2) identify all safety-related equipment that was tested to single-axis or single-frequency, (3) review the qualification documents for compatibility to USNRC NUREG-0800, SRP Section 3.10, II-2 requirements (Reference 5), (4) perform remedial action for any identified deficiencies, and (5) document results.

As part of Step 3, the Work Plan specifies that the review of the qualification documents: (1) ensures sufficient test input conservatism to account for directional coupling and multi-mode effects, (2) TRS envelopes the RRS in all directions, (3) evaluates the impact of relay contact chatter to its specific application, and (4) evaluates the impact of equipment structural changes.

To ensure sufficient test input conservatism for directional coupling and multi-mode effects, the Work Plan calls for the development of technical guidelines. A copy of the draft Guidelines was provided with the Work Plan. The guidelines specify that existing qualification documents be reviewed to develop justification for directional coupling and multi-mode effects in accordance with TVA Design Criteria No. WB-DC-40-31.2 (Reference 4) methods. According to the guidelines, those methods are in agreement with the SRP Section 3.10, II.1.a(5) (Reference 5) and IEEE Standard 344-1975, Section 6.6.2 (Reference 6).

A review of the design criteria demonstrated that it correctly specifies the conditions when single-axis or single-frequency testing or both is acceptable. For example, single-axis testing is allowed if the equipment being tested can be shown to respond independently in each of the three orthogonal axes. An example in the design criteria for adequacy of single-frequency testing is when the seismic ground motion has been filtered due to one predominant structural mode. This is characterized by a single, sharp, narrow-band floor response spectrum. Since the methods for justification of single-axis and single-frequency testing specified in the design criterion are in accordance with SRP Section 3.10 and IEEE 344-1975, they are acceptable.

If one of the methods described in the TVA design criteria cannot provide justification, then the guidelines specify that qualification documentation of similar equipment qualified for other sites, where directional coupling and multi-mode effects have been addressed, can be used.

If such efforts are still not successful, then the guidelines require that a modal analysis be performed to develop a cross-coupling multiplication factor to be used in the comparison of TRS to RRS. For multi-mode effects, the guidelines refer to Section 6.6.2.1 of Reference 6 to develop the multiplication factor to account for single-frequency tests. The product of the two multiplication factors and the RRS is then used to compare against the TRS.

The guidelines also define other alternatives such as in situ tests or modification of equipment (e.g., stiffening of cabinet) to address the concerns with single-frequency and single-axis testing.

To address the IFI 90-05-02 concern regarding relay chatter, TVA has addressed this concern in Step 3 of the work plan described above. It requires that any relay chatter be evaluated for acceptability on the basis of the specific application of the relay. This will be performed for all safety-related relays supplied by Westinghouse, whether above or below 0.95g.

In addition to the Work Plan, Section 3.2.2 of TVA's Design Criteria (Reference 4) describes seismic qualification requirements for such devices as relays. It states that a device, as an integral part of an assembly (e.g., panel), can be subjected to seismic tests while in an operating condition and its performance can be monitored during the test. Alternatively, the individual devices are tested separately in an operating condition and the test levels are recorded as the qualification levels of the devices. The panel, with similar devices installed but inoperative, is vibration tested to determine panel response accelerations. If the acceleration levels at the device locations are found to be less than the levels to which the device has been qualified, then the total assembly may be considered qualified. This approach is consistent with requirements specified in IEEE 344-1975 (Reference 6) and thus is acceptable.

TVA's proposed action plan is in accordance with the USNRC SRP Section 3.10 and IEEE 344-1975. On this basis, IFI 90-05-02 is considered closed.

<u>References</u>


1. "Watts Bar Safety Evaluation Report," NUREG-0847, Supplement No. 3, January 1985.

2. USNRC Inspection Report 50-390, 391/90-05, Watts Bar Nuclear Plant, Units 1 and 2, May 1990.

3. TVA internal document, "NRC Audit, Equipment Seismic Qualification, 24-28 February 1992," RIMS No. T30 920227 827.

4. TVA Design Criteria No. WB-DC-40-31.2, "Seismic Qualification of Category 1 Fluid System Components and Electrical or Mechanical Equipment," Rev. 5.

5. USNRC NUREG-0800, Standard Review Plan 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment."

6. IEEE Standard 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."


## 5. NEW ISSUES

No new issues were identified as a consequence of this audit. However, the concern regarding the seismic adequacy of the cable tray cover and its attachment remains. This concern is being tracked under URI 88-01-02.

## 6. LIST OF PERSONS CONTACTED

W. A. Massie*, Licensing, TVA/WBN
R. Hernandez*, Project Engineer, TVA/NE
C. Y. Chiou*, Supervising Engineer, Ebasco
G. L. Pannell*, Site Licensing Manager, TVA
J. Garrity*, Site VP, TVA
J. Sledje*, Task Manager, MODS
J. Hawkins*, QA Specialist, QA
R.L. Cloud*, Consultant, RLCA
J.A. Adair*, Lead Civil Manager, TVA
J. Chen, TVA
T. Cureton, TVA
M. McGrath, Ebasco
J. Aros, EQE
T. Kipp, EQE

* Attended exit meeting

# APPENDIX T

## SUPPLEMENTAL SAFETY EVALUATION*: CORRECTIVE ACTION PROGRAM FOR CABLE ISSUES

*Originally issued by letter, P. S. Tam to M. O. Medford, dated April 24, 1992. The safety evaluation on corrective action for cable issues was issued as Appendix P of SSER 7.

## SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## WATTS BAR NUCLEAR PLANT, UNIT 1

## DOCKET NO. 50-390

## CORRECTIVE ACTION PROGRAM FOR CABLE ISSUES

## METHODOLOGY FOR ELECTRICAL CABLE - HOT PIPE SEPARATION

### INTRODUCTION

In the safety evaluation on the cable issues corrective action program of April 25, 1991, the staff mentioned the issue of cable proximity to hot pipes. At that time, the staff had not yet evaluated the applicant's mathematical model used for this issue.

Ebasco Services Incorporated, under contract to TVA, is performing calculations and analyses to establish acceptable clearances between electrical cables and hot pipes. Overheating of the cables could lead to premature aging which could cause eventual failure of the cable insulation. Electrical cables within proximity to heated pipes may be affected by convection plumes of heated air, as well as by thermal radiation. Internal electric resistance heating must also be considered. The calculational methods involve a number of computer codes developed by Ebasco to calculate radiation and convection heat transfer from component surfaces and simple conduction within a cable tray. For complex conduction analyses in three dimensions, Ebasco uses the HEATING5 computer code developed by the Oak Ridge National Laboratory (ORNL).

### EVALUATION OF SPECIFIC METHODS

This issue was discussed in a site audit during the week of March 2, 1992. Among other things, the staff reviewed Watts Bar site calculation WBN-OSG4-138, "Class 1E Electrical Cable/Hot Pipe Clearance Requirements". The staff has reviewed the more significant computer codes and subroutines used by the applicant. Findings are described below:

CTRHT - The heat conduction equation is solved to find the maximum temperature within a bundle of cables within a cable tray. Internal electric heating within the cable bundle is considered. Heat transfer is assumed only from the top and bottom bundle surfaces of the cable tray which is conservative and makes the problem one-dimensional. Heat transfer boundary conditions at the top and bottom surfaces are determined elsewhere.

HTCNC - Natural convection heat transfer coefficients are determined for vertical and horizontal plates and cylinders. Correlations of experimental data are used as they appear in standard texts. The resulting coefficients are functions of the product of the Grashof and Prandtl numbers. The staff made audit checks of the correlations programmed into the computer code and found the results to be acceptable.

QRAD - Radiation heat transfer between a surface exposed to one hot surface and background is calculated. The geometrical shape factors and emissivities are used as input. The Stefan-Boltzmann equation modified for gray body radiations used according to standard engineering practice.

CONDHT - The surface temperature of conduits containing cables is calculated. The calculations consider radiation heat transfer from an adjacent hot pipe (QRAD), natural convection heat transfer (HTCNC), the effect of a heated plume from a hot pipe below the conduit (THPLUM) and internal heating from up to 24 electric load bearing cables. The program iterates on conduit/pipe spacing until an acceptable conduit surface temperature is obtained. Internal cable temperatures are not calculated directly but are accounted for by using electrical design standards. The calculation is made conservative by using the heat flux for the smallest conduit that can hold the specified number of cables in accordance with the design standard.

PTEMP80 - This program calculates the surface temperature of Mirror insulation covering hot pipes within the containment building. The calculation is based on a design commitment by the manufacturer to limit surface heat loss to 80 BTU/hour/square foot. Surface heat transfer is calculated by the HTCNC computer program for convective heat transfer and QRAD for radiative heat loss. The method should be accurate provided that the Mirror insulation is installed in accordance to the manufacturer's commitment to limit heat flow to 80 BTU/hour/square foot.

Radiation shape factors - These refer to the fraction of thermal radiation leaving one heated object which then strikes another body. The shape factors were derived by digital integration of the conduit and hot pipe surfaces using standard formulas.

Convection heating - Temperatures of cables inside conduits or cable trays will increase when located within thermal plumes that rise from hot pipes. The temperature, velocity and width of rising thermal plumes are calculated by the THPLUM, VPTLA and HORIZN computer programs. The effect of heated plumes is considered in determining the heat transfer and bulk temperature boundary conditions.

HEATING5 - This is a three-dimensional heat conduction computer program developed by the Oak Ridge National Laboratory which is described in ORNL/CSD/TM-15 dated March 1977. HEATING5 has been successfully used by the NRC staff and its contractors to calculate temperature distributions that might occur in the long-term storage of high level nuclear waste. With proper input, the code would accurately calculate temperature distributions within electrical conduits and cable trays.

Calculational results - Based on computer analyses, Ebasco arrived at the following general guideline for parallel runs of insulated pipes and cables, while other guidelines are being developed for asymmetric configurations and uninsulated pipes: (1) Conduits and cable trays must maintain a clearance of

6 inches from an insulated hot pipe when run in parallel to the pipe but not above it. (2) Conduits run parallel, and above an insulated hot pipe must maintain a clearance of 1.5 times the outer diameter of the pipe insulation; cable trays in this configuration must maintain a clearance of 0.5 times the outer diameter of the pipe.

CONCLUSION

The staff reviewed the models and assumptions used at Watts Bar to determine electrical cable temperatures, and has determined that the models and assumptions are reasonable for calculations of this type. All relevant physical phenomena appear to have been considered. The detailed temperature results from the computer analyses were not submitted and were therefore not reviewed. The staff understands that TVA plans to verify these calculations during hot functional testing by performing temperature surveys of locations that might be subject to overheating.

Principal contributor: W. Jensen

Dated: April 1992

APPENDIX U

SUPPLEMENTAL SAFETY EVALUATION: CORRECTIVE ACTION PROGRAM ON MASTER FUSE LIST

*Originally issued by letter, P. S. Tam to Senior Vice President, dated March 30, 1992.

SUPPLEMENTAL SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING THE WATTS BAR CORRECTIVE ACTION PROGRAM

ON THE MASTER FUSE LIST

## INTRODUCTION

In Section 3.3.5 of NUREG-1232, Vol. 4, "Safety Evaluation on Tennessee Valley Authority: Watts Bar Nuclear Performance Plan", the staff concluded that the Tennessee Valley Authority's (TVA's) Master Fuse List (MFL) program has identified the root cause of fuse misapplication and that the corrective action is adequate. However, in the SE, the staff also indicated that TVA had not submitted documentation to resolve the three staff concerns discussed below and that TVA should resolve these concerns.

(1) Fuse sizes should be removed from drawings and be replaced with an identifier number that also appears on the MFL. This reduces errors on the drawings and provides a complete design description on one document which is the MFL.

(2) Installed fuses are verified to agree with the MFL. Although TVA has agreed to replace any fuse that does not agree with the MFL, there is no procedure that requires a walkdown to verify that all the installed fuses agree with the MFL.

(3) Adequate administrative controls are in place to ensure that after the verified walkdown, any fuse replacement agrees with the MFL.

By letters dated July 31, 1990, May 31, 1991, and January 31, 1992, TVA responded to each of these concerns. The staff's evaluation for each of these responses is described as follows:

## EVALUATION

### Concern (1)

The Master Fuse List program, described in Chapter III, Section 3.5 of the Watts Bar Nuclear Performance Plan, indicated that fuse sizes that currently appear on design drawings would also be included on a Master Fuse List that is being developed. The staff expressed the concern that the size of fuses should be removed from drawings and should be replaced with an identifier number that also appears on the Master Fuse List. The staff felt that this removal and replacement of fuse sizes with an identifier number would reduce errors on drawings and would provide a complete design description on one document, which is the Master Fuse List.

Based on information provided by letter dated July 31, 1990, in response to this staff concern, it appeared that fuse sizes would continue to be included on both design drawings and the Master Fuse List, but a note would be added to each drawing stating that for information regarding Class 1E fuses, refer to the Master Fuse List drawing series 45B6000 for configuration control of these fuses. It was not clear how this note would resolve the staff concerns for reducing errors between the Master Fuse List and design drawings, and how one could establish collation between the Master Fuse List and fuses shown on design drawings.

Additional information provided by letter dated May 31, 1991, and conveyed during a September 4, 1991, telephone conference call, indicated that TVA would reconcile the Master Fuse List with other TVA design output drawings to ensure consistency. The fuse sizes would be verified by either reconciling them on the drawings and the Master Fuse List or by eliminating the fuse sizes from the drawings (leaving only the unique fuse identifier) and providing a note on the drawings to obtain applicable fuse information from the Master Fuse List.

Based on the commitment to reconcile the Master Fuse List with other TVA design output drawings to ensure consistency, the staff concluded that there would be reasonable assurance that discrepancies between the Master Fuse List and design output drawings would be initially eliminated. The commitment to reconcile is therefore acceptable. However, if the fuse sizes were left on the drawing after they are reconciled with the Master Fuse List (as indicated above) and there is a subsequent modification which causes the Master Fuse List to change, discrepancies between the Master Fuse List and design output drawings may be reintroduced. The staff concern therefore remained.

Additional information provided by letter dated January 31, 1992, indicates that TVA will remove fuse sizes from drawings prior to fuel load for fuses listed on the MFL. Based on TVA's commitment to remove fuse sizes from drawings and to add a note on drawings referring to the MFL for fuse information, the staff considers this concern acceptably resolved.

Concern (2)

The Master Fuse List program, described in Chapter III, Section 3.5 of the Watts Bar Nuclear Performance Plan, implies that the Master Fuse List was developed through an engineering analysis of circuits. When an incorrect type or size of fuse was established, the Master Fuse List was changed to indicate the correct fuse. After the engineering analysis of circuits was completed and the Master Fuse List updated, the staff was concerned that there would be no procedure or other methodology that requires a walkdown to verify that all the installed fuses agree with the Master Fuse List.

In response to this concern, TVA stated in the July 31, 1990, letter that information on fuses on the Master Fuse List has been verified using walkdown information. Based on this statement, it appeared that the staff concern may have been acceptably resolved. However, in further response to this concern, TVA provided a commitment to verify the installed configuration using the Master Fuse List only if walkdown data was not obtained and if fuses were found missing during the walkdown inspection. This commitment was acceptable, but did not resolve the staff concern. This commitment implied that the walkdown inspection was performed prior to the engineering analysis and that the Master Fuse List had been extensively changed (subsequent to the walkdown inspection) as a result of the engineering analysis. It was thus not clear how or by what procedure a fuse would be replaced when the fuse type or size specified in the Master Fuse List is changed. It was also not clear how existing fuses will be verified to agree with the Master Fuse List prior to fuel load.

Additional information provided by letter dated May 31, 1991, indicated:

1. After the Master Fuse List is reconciled with other design output drawings and with the installed configuration, future fuse design changes will be controlled by the Watts Bar Engineering Procedure (WBEP) 5.03, "Design Change Notices." The Master Fuse List is a design output document controlled by this process. The design change notice will be implemented by site administrative instructions.

2. TVA will perform a walkdown to ensure that those fuses depicted on the Master Fuse List agree with the plant's installed configuration.

3. It should be noted that the Master Fuse List does not contain all Class 1E fuses. The focus of the Master Fuse List Program is to support operations by depicting fuse information for those Class 1E safety-related fuses which plant operators typically change. Other fuses requiring change-out are procedurally controlled by the work order or work plan process which requires documentation/justification of the replacement.

   The Master Fuse List contains the following categories of Class 1E fuses that are depicted on the TVA-controlled design drawings.

   -- Class 1E fuses used in the auxiliary power and control systems,
   -- Class 1E fuses used as electrical penetration protection fuses,
   -- Class 1E fuses used in control/distribution panels, and
   -- Class 1E fuses that perform 1E to non-1E isolation functions.

The above information clarified and thus resolved by what procedure a fuse will be replaced when the fuse type or size specified in the Master Fuse List is changed, and how fuses that have been included on the Master Fuse List will be verified to be consistent with the design output drawings. For fuses identified on the Master Fuse List, TVA's commitment to perform a walkdown to ensure that those fuses depicted on the Master Fuse List agree with the plant's installed configuration partially resolved the staff concern. As indicated in item 3 of the above information provided by TVA, all Class 1E fuses will not be included on the Master Fuse List. For Class 1E fuses not included on the Master Fuse List, our original concern remained.

In order to resolve the staff concern for fuses that will not be included on the Master Fuse List, additional information was required which justifies not including some of the Class 1E fuses on the Master Fuse List. TVA's letter dated January 31, 1992, stated that Class 1E fuses not on the MFL are in vendor-supplied design packages and controlled by the vendors' quality assurance programs. Also, these fuses are for the protection of specific vendor-supply equipment and not required for overall electrical system coordination. TVA committed to annotate the MFL with a note that directs users to vendor documents for those Class 1E fuses not specifically listed. Based on this commitment, the staff considers this concern acceptably resolved.

Concern (3)

In its July 31, 1990, response, TVA indicated that administrative instruction for fuse control has been revised to ensure that the fuse configuration agrees with the Master Fuse List for all Class 1E safety-related fuses. Based on this response, the staff concluded that adequate administrative controls would be in place to ensure that after any fuse replacement, the new fuse will agree with the Master Fuse List. This concern was therefore considered acceptably resolved for fuses on the Master Fuse List.

In order to resolve the staff concern for fuses that will not be included on the Master Fuse List (see Concern 2 above), additional information was required which addresses the administrative procedures which will assure correct fuse replacement for the fuses that are not on the Master Fuse List. In response, TVA's letter dated January 31, 1992, stated that plant personnel either replace those fuses with ones having the same manufacturer and model numbers or use the vendor manuals to determine appropriate fuse selection. Based on this response, the staff considers this concern adequately resolved.

CONCLUSION

As stated above, TVA has acceptably resolved the staff's three concerns communicated in NUREG-1232, Vol. 4. This completes the program review of the Corrective Action Program on the Master Fuse List.

Principal contributor: Fred Burrows

Dated: March 1992

EGG-NTA-9563

# TECHNICAL EVALUATION REPORT

## CONFORMANCE TO REGULATORY GUIDE 1.97: WATTS BAR-1/-2

Docket Nos. 50-390/50-391

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## SUMMARY

This report documents the EG&G Idaho, Inc., review of the Watts Bar Nuclear Plant submittals regarding Regulatory Guide 1.97, Revision 2. This report identifies areas of non-conformance to the regulatory guide. Exceptions to Regulatory Guide 1.97 are evaluated. The applicant either conforms to or is justified in deviating from the guidance of Regulatory Guide 1.97 for each variable.



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## PREFACE

This report is supplied as part of the "Program for Evaluating Licensee/Applicant Conformance to RG 1.97." It is being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Systems Technology, by EG&G Idaho, Inc., Regulatory and Technical Assistance Unit.

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CONFORMANCE TO REGULATORY GUIDE 1.97: WATTS BAR-1/-2

## 1. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 (Reference 1) was issued by D. G. Eisenhut, Director of the Division of Licensing, Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits. This letter included additional clarification regarding Regulatory Guide 1.97, Revision 2 (Reference 2), relating to the requirements for emergency response capability. These requirements have been published as Supplement No. 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

Tennessee Valley Authority, the applicant for the Watts Bar Nuclear Plant, responsed to Item 6.2 of the generic letter on January 30, 1984 (Reference 4). This response reviewed the post-accident monitoring instrumentation provided to the recommendations of Regulatory Guide 1.97.

The applicant provided additional information on April 16, 1985 (Reference 5). The applicant provided an additional submittal on August 31, 1990 (Reference 6). The Reference 6 submittal completely replaced the earlier submittals. This information was supplemented on October 11, 1990 (Reference 7), and January 3, 1991 (Reference 8), with schedular information. The applicant provided additional information on October 29, 1991 (Reference 9).

This report is based on the recommendations of Regulatory Guide 1.97, Revision 2. This report compares the instrumentation proposed by the applicant's August 31, 1990, submittal, as supplemented by schedular submittals of October 11, 1990, and January 3, 1991, and additional information dated October 29, 1991, with these recommendations.

## 2. REVIEW REQUIREMENTS

Item 6.2 of NUREG-0737, Supplement No. 1, lists the documentation to be submitted in a report to the NRC describing how the applicant complies with Regulatory Guide 1.97 as applied to emergency response facilities. The documentation should provide the following information for each variable shown in the applicable table of Regulatory Guide 1.97.

1. instrument range
2. environmental qualification
3. seismic qualification
4. quality assurance
5. redundance and sensor location
6. power supply
7. location of display
8. schedule of installation or upgrade

The submittals should identify any deviations taken from the regulatory guide recommendations and provide supporting justification or alternatives for the deviations identified.

Subsequent to issuing the generic letter, the NRC held regional meetings, in February and March 1983, to answer licensee and applicant questions and concerns regarding the NRC policy on this subject. At these meetings, it was noted that the NRC review would address only the exceptions taken to Regulatory Guide 1.97. It was also noted that when licensees or applicants explicitly state that instrument systems conform to the regulatory guide, no further staff review would be necessary. Therefore, this report addresses only those exceptions to Regulatory Guide 1.97 identified by the applicant. The following evaluation is an audit of the applicant's submittals based on the review policy described in the NRC regional meetings.

## 3. EVALUATION

The applicant provided responses to Item 6.2 of NRC Generic Letter 82-33 on January 30, 1984, April 16, 1985, August 31, 1990, October 11, 1990, January 3, 1991, and October 29, 1991. The August 31, 1990 (Reference 6), response combines and replaces the previous responses. It describes the applicant's position on post-accident monitoring instrumentation. The October 11, 1990, and January 3, 1991, submittals provide schedular information. The October 29, 1991, submittal clarifies and embellishes the earlier submittals. This evaluation compares the material provided in Reference 6 and later submittals to the recommendations of Revision 2 of Regulatory Guide 1.97.

### 3.1 Adherence to Regulatory Guide 1.97

The applicant documented their review of the Watts Bar Nuclear Plant post-accident monitoring instrumentation. The applicant based their review on Revision 2 of Regulatory Guide 1.97. The review included the Watts Bar emergency instructions and a safety analysis incorporating the Final Safety Analysis Report (FSAR) Chapter 15 design basis accidents. The applicant provided justification where deviations exist from the recommendations of the regulatory guide. The applicant discussed the design criteria used for the instrumentation involved. The applicant scheduled upgrades to their instrumentation. Therefore, we conclude that the applicant has provided an explicit commitment on conformance to Regulatory Guide 1.97. Exceptions to and deviations from the regulatory guide are noted in Section 3.3.

### 3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, that is, those variables that provide the information required to permit the control room operator to take specific, manually-controlled safety actions. The applicant classifies the following instrumentation as Type A.

1. auxiliary feedwater flow
2. containment lower compartment atmospheric temperature
3. containment pressure (-2 psig to 15 psig)
4. containment radiation
5. containment sump water level (zero to 20 feet)
6. core exit temperature
7. main steamline radiation
8. neutron flux -- source range (1 to $10^6$ counts per second)
9. reactor coolant system (RCS) pressurizer level
10. RCS pressure (zero to 3000 psig)
11. RCS cold leg water temperature
12. RCS hot leg water temperature
13. refueling water storage tank level
14. steam generator level -- narrow range
15. steam generator pressure
16. subcooling margin monitor

These variables, with exceptions as noted in Section 3.3, either meet, or the applicant will modify the instrumentation to meet, the Category 1 recommendations, consistent with the requirements for Type A variables.

### 3.3 Exceptions to Regulatory Guide 1.97

The applicant identified deviations and exceptions to Regulatory Guide 1.97. The following paragraphs discuss these deviations and exceptions.

#### 3.3.1 RCS Soluble Boron Concentration

Regulatory Guide 1.97 recommends Category 3 instrumentation for this variable with a range of zero to 6000 parts per million. The applicant has not provided on-line instrumentation for this variable. Instead, the applicant samples with the post-accident sampling facility. The tolerance for a reading of boron concentration less than 500 parts per million is ±50 parts per million. With this uncertainty, the lower limit of the recommended range may not be realistically attainable.

The applicant deviates from the recommendations of Regulatory Guide 1.97 for post-accident sampling capability. This deviation goes beyond the scope of this review and was approved by the NRC as part of their review of NUREG-0737, Item II.B.3.

#### 3.3.2 RCS Cold Leg Water Temperature

Revision 2 of Regulatory Guide 1.97 recommends a range of 50°F to 750°F for this variable. The applicant has instrumentation with a range of 50°F to 700°F. The applicant states that 650°F is the design temperature of the RCS and that the maximum temperature during any expected transient is less than 700°F. The provided range allows a 50°F margin over the design temperature to accommodate transients.

The applicant shows the 50°F to 700°F range exceeds all expected design basis conditions. Further, Revision 3 of Regulatory Guide 1.97 (Reference 10) recommends a range of 50°F to 700°F. The applicant's instrumentation meets this range. Based on this, we find this deviation acceptable.

### 3.3.3 RCS Hot Leg Water Temperature

Revision 2 of Regulatory Guide 1.97 recommends a range of 50°F to 750°F for this variable. The applicant has instrumentation with a range of 50°F to 700°F. The applicant states that 650°F is the design temperature of the RCS and that the maximum temperature during any expected transient is less than 700°F. The provided range allows a 50°F margin over the design temperature to accommodate transients.

The applicant shows the 50°F to 700°F range exceeds all expected design basis conditions. Further, Revision 3 of Regulatory Guide 1.97 (Reference 10) recommends a range of 50°F to 700°F. The applicant's instrumentation meets this range. Based on this, we find this deviation acceptable.

### 3.3.4 Containment Sump Water Level -- Narrow-Range

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable with a site specific range. The applicant has Category 3 instrumentation for the reactor building sump with a range of 2 inches to 66 inches.

The applicant states that the leakage rate before an accident is the parameter of concern monitored by this instrumentation. The instrumentation is not the basis for operator or automatic operation of safety-related equipment. The leakage monitored is not enough to cause plant perturbations or transients that would cause a reactor trip signal or a safety-injection signal.

We conclude that the applicant has provided appropriate monitoring for the parameter of concern. We base this conclusion on the following.

a. For small leaks, the instrumentation will not experience harsh environments during operation and will show response to the leak.

b. For larger leaks, the sumps fill promptly. The sump drain lines isolate due to the increase in containment pressure. This negates the narrow-range instrumentation. The wide-range instrumentation will still provide valid signals.

c. This instrumentation neither automatically starts nor alerts the operator to start operation of a safety-related system in a post-accident situation.

Therefore, we find the Category 3 narrow-range instrumentation provided acceptable.

### 3.3.5 Containment Pressure

Regulatory Guide 1.97 recommends readouts of zero to design pressure, 10 psia to design pressure, and 10 psia to 4 times design pressure for this variable. The applicant has instrumentation with readouts of -2 psig to 15 psig and -5 psig to 60 psig. The design pressure is 15 psig for this ice condenser containment design. The applicant identified a deviation in the range of the narrow-range instruments (-2 psig to 15 psig).

The applicant's wide-range instruments cover the negative pressures recommended by the regulatory guide with no discernable loss of resolution. Therefore, the applicant's instrumentation is acceptable.

### 3.3.6 Containment Isolation Valve Position

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. The applicant identified a deviation from this recommendation. Flow control valves (FCV) that are part of the RCS letdown system will become submerged following an accident. The valves effected are FCV-62-72, FCV-62-73, FCV-62-74, and FCV-62-76. These valves have limit switches not qualified for submerged operation.

The applicant states that the instrumentation shows the valve position well in advance of valve submergence. The control power fuse is assumed to open when water enters the valve limit switch enclosure. This loss of power causes the loss of position indication. The applicant states the valves will not change position after submergence because the valves vent closed on loss of control power. Testing verified this design. Based on the described operation of the valves, we find this deviation acceptable.

### 3.3.7 Radiation Level in Circulating Primary Coolant

The applicant states that the post-accident sampling facility provides radiation level measurements to show fuel cladding failure. The NRC reviewed and approved this capability as part of their review of NUREG-0737, Item II.B.3.

Based on the alternate instrumentation provided by the applicant, we conclude that the instrumentation supplied for this variable is adequate and, therefore, acceptable.

### 3.3.8 Containment Hydrogen Concentration

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of zero to 30 percent for an ice condenser containment. The applicant's instrumentation has a range of zero to 10 percent. The applicant performed an analysis showing the worst case containment hydrogen concentration will be less than 4 percent with one of the hydrogen recombiners operating. The hydrogen igniter system, separate from the hydrogen recombiners, also maintains the hydrogen concentration at less than 10 percent. The control room has indication of the hydrogen recombiner system operation. Additionally, the post-accident sampling facility can provide a diverse source of information concerning containment hydrogen concentration. The NRC approved a range to 10 percent for this system. See Section 3.3.33.

As this instrumentation remains on-scale for all post-accident situations, we find the zero to 10 percent range provided for this instrumentation acceptable.

### 3.3.9 Radiation Exposure Rate

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable with a range of $10^{-1}$ R/hr to $10^{4}$ R/hr. This instrumentation shows containment breach, detects significant releases, aids in release assessment, and enables long-term surveillance. Revision 3 of the regulatory guide deletes the instrumentation for measuring releases caused by a containment breach.

The applicant identified instrumentation to monitor the radiation level in the main control room. However, the range identified is $10^{-1}$ mR/hr to $10^{4}$ mR/hr. The limits of the provided range are three decades less than recommended. However, the applicant states that the maximum expected radiation level in this continually manned area is 100 mR/hr. As the instrumentation will remain on-scale for all expected conditions, we find this deviation in range acceptable.

The applicant has instruments with ranges of $10^{-1}$ mR/hr to $10^{4}$ mR/hr located throughout the plant. The applicant states that those monitors located outside of primary containment will remain on-scale with the required accuracy during accident conditions. For primary containment entry, the applicant uses portable instruments with a range of $10^{-3}$ R/hr to $10^{4}$ R/hr.

From a radiological standpoint, should radiation levels reach or exceed the provided upper limit, $10^{4}$ mR/hr, personnel would not enter these areas except for lifesaving. We therefore find the applicant's range ($10^{-1}$ mR/hr to $10^{4}$ mR/hr) for the radiation exposure rate monitors acceptable.

The applicant identified Category 3 instrumentation for this variable. As Revision 3 of Regulatory Guide 1.97 (Reference 10) accepts Category 3 instrumentation for this variable, we find the applicant's instrumentation for this variable acceptable.

### 3.3.10 RHR Heat Exchanger Outlet Temperature

Regulatory Guide 1.97 recommends a range of 32°F to 350°F for this variable. The applicant's instrumentation has a range of 50°F to 400°F.

Revision 3 of the regulatory guide (Reference 10) increases the minimum recommended range to 40°F. Thus the lower limit of the range deviates from the recommended limit by 10°F. Considering instrument accuracy, we find this deviation minor and acceptable.

### 3.3.11 Accumulator Tank Level and Pressure

Regulatory Guide 1.97 recommends Category 2 instrumentation with ranges of zero to 750 psig and 10 percent to 90 percent volume. Watts Bar has Category 3 pressure instrumentation for these accumulators with a range of zero to 700 psig. This range exceeds the technical specification requirements that require a nitrogen blanket pressure of 632 psig. A pressure relief valve maintains pressure below 700 psig. We find the present zero to 700 psig range acceptable. This range will enable the operator to see that the accumulators have discharged.

The applicant identified a range of 75 percent to 82 percent for the Category 3 level instrumentation. This range monitors the level of the accumulators during normal operation. We note that the range of the level instrumentation coincides with the technical specification requirements to maintain between 7,632 gallons and 8,264 gallons in each accumulator.

The applicant states that this instrumentation primarily shows the pre-accident status of the accumulators. The applicant also has separate, independent Category 3 low pressure alarms and low level alarms for each accumulator.

The accumulators contain borated water and a blanket of pressurized nitrogen gas. The accumulators are a passive system. They provide a fast-acting, high flow rate injection. The flow is into the cold legs of the reactor coolant system during the injection phase of a large loss of coolant accident recovery. The accumulators are isolated from the reactor coolant system during normal reactor operation by two series-connected check valves. Each accumulator also has a motor-operated isolation valve. These valves are open before power operation. Once open, their power is removed. Technical Specification surveillance requirements assure the status of these isolation valves. Should the reactor coolant system pressure decrease below the accumulator pressure, the check valves open. This allows the blanket nitrogen gas pressure to force the borated water into the reactor coolant system. The mechanical operation of the swing-disk check valves is the only action required for this injection. No external power source, initiating signal, or post-accident monitoring instrumentation is necessary for the accumulators to perform their safety function. The operator must take deliberate actions to close the isolation valves. Post-accident isolation of the accumulators from the reactor coolant system is not a safety function. Once the accumulators have automatically discharged, they have no further post-accident safety function. The level instrumentation will read off-scale (low) following a discharge and accumulator discharge is verifiable by the pressure instrumentation. Therefore, we find the range of the level instrumentation acceptable.

Because of the design basis of the accumulator operation, the accumulator instrumentation does not perform a safety function during post-accident recovery. Thus, the instrumentation does not perform a safety function in a post-accident environment. Reactor operators do not depend on the accumulator instrumentation to decide on actions necessary to mitigate the effects of an accident. Emergency core cooling systems operate independently of the accumulators. The emergency core cooling system performance can be observed on other Category 1 and 2 instrumentation.

We conclude that the post-accident monitoring of the accumulator level and pressure does not perform a safety function. There are no operator actions based on the accumulator level and pressure instrumentation requiring Category 2 instrumentation. Therefore, we find the use of Category 3 instrumentation to monitor the accumulator level and pressure acceptable.

### 3.3.12 Accumulator Isolation Valve Position

Regulatory Guide 1.97 recommends Category 2 indication of the position of these valves. The applicant states that these valves do not change position following an accident. By design, the operator manually removes power to these valves as part of the startup procedure. Because these valves are open and cannot inadvertently change position during or following an accident, we consider the Category 3 indicators for the position of these valves acceptable.

### 3.3.13 Boric Acid Charging Flow

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The applicant's boric acid charging flow instrumentation is Category 3.

The applicant states that this flow path is a normally isolated line and requires manual operator action for use. It is not one of the safety injection paths at Watts Bar. Based on this design, we find the instrumentation provided for this variable acceptable.

3.3.14 <u>Pressurizer Heater Status</u>

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The applicant states, in Reference 6, they have provided the appropriate instruments for two safety-related banks of heaters in the Technical Support Building. Reference 9 corrects the statement, stating the display is in the main control room. These heater banks can be loaded onto their associated diesel-generator. We find the instrumentation provided for the safety-related heater banks acceptable for this variable.

3.3.15 <u>Quench Tank (Pressurizer Relief Tank) Temperature</u>

Regulatory Guide 1.97 recommends a range of 50°F to 750°F for this variable. The applicant's instrumentation has a range of 50°F to 300°F. The regulatory guide states that the instrumentation should always be on-scale. The pressurizer relief tank rupture disc operates at 93 psig ±7 psig. Assuming worst case limits, this pressure relief limits the temperature of the tank contents to saturated conditions, about 350°F. The applicant committed to expand the range limits to 50°F and 400°F (Reference 9). This change includes the saturated steam conditions (approximately 350°F) corresponding to 100 psig. We find this commitment acceptable.

3.3.16 <u>Steam Generator Level -- Wide-range</u>

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. The applicant, in Reference 6, committed to provide one Category 1 instrument channel per steam generator. Regulatory Guide 1.97 specifically allows one channel of this instrumentation per steam generator. Thus, the applicant's instrumentation for this variable will be in conformance to Regulatory Guide 1.97.

### 3.3.17 Steam Generator Pressure

Regulatory Guide 1.97 recommends instrumentation for this variable with a range from zero to 20 percent above the lowest safety valve setting. The applicant's instrumentation has a range of zero to 1300 psig. The range corresponding to the regulatory guide recommendation would be zero to 1422 psig.

The applicant provides the following justification for using the zero to 1300 psig range. First, the design pressure of the main steam system is 1185 psig. Second, all main steam safety valves are open at 1224 psig or below. With all the main steam safety valves open, the pressure is, by design, not expected to exceed 1284 psig.

Based on the applicant's justification, we find that the main steam system pressure will not exceed the zero to 1300 psig range of the pressure instrumentation. Therefore, we find the provided range acceptable.

### 3.3.18 Auxiliary Feedwater Flow

Regulatory Guide 1.97 recommends a range of zero to 110 percent of design flow. The applicant states that each of the four auxiliary feedwater loops has two channels of flow instrumentation. The applicant states that each channel covers to at least 110 percent of the design flow (Reference 9) at the transmitter location. Based on this statement, we find the range provided acceptable.

### 3.3.19 Condensate Storage Tank Water Level

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable except where the condensate storage tank is not the primary source of auxiliary feedwater. Should another source be the primary source of auxiliary feedwater, the applicant should identify it and use Category 1 instrumentation to monitor it. The applicant has Category 3 instrumentation for the condensate storage tank water level. Category 3 instrumentation is acceptable for the condensate storage tank water level instrumentation when using another source of auxiliary feedwater.

The applicant states that the essential raw cooling water system is the primary source of auxiliary feedwater (Reference 9). The applicant checks the auxiliary feedwater valves (essential raw cooling water interface) position with Category 1 instrumentation. With this compliment of instrumentation, the applicant satisfies the Regulatory Guide 1.97 recommendations for this variable.

### 3.3.20 Heat Removal by Containment Fan Heat Removal System

Regulatory Guide 1.97 recommends plant specific Category 2 instrumentation for this variable. The applicant states the ice condenser containment design does not require qualified lower containment ventilation units for accident recovery. Based on this design, the Category 3 containment cooling water valve status instrumentation and the Category 2 fan status instrumentation is acceptable for this variable.

### 3.3.21 Containment Atmosphere Temperature

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 40°F to 400°F. The instrumentation provided by the applicant for this variable has a range of zero to 350°F.

The applicant states that the maximum post-accident temperature expected follows a steamline break. This peak containment temperature is 327°F. Based on this limit, we find the zero to 350°F range acceptable.

### 3.3.22 Containment Sump Water Temperature

The applicant uses alternate instrumentation for this variable. In the recirculation mode, the residual heat removal pumps take suction from the containment sump. The pumps discharge into the residual heat removal heat exchangers. The sump temperature is available in the control room by checking the RHR heat exchanger inlet temperature. Category 2 instrumentation monitors this temperature.

The water temperature at the residual heat removal heat exchanger inlet will be the same as the sump temperature. Therefore, we find this alternate instrumentation acceptable.

### 3.3.23 Makeup Flow-In

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The applicant's instrumentation is Category 3. The applicant states that the chemical and volume control system makeup is the normal charging flow path. The chemical and volume control system automatically isolates with a safety injection signal.

Regulatory Guide 1.97 recommends this instrumentation for monitoring the chemical and volume control system. The information provided shows the chemical and volume control system isolates post-accident. Therefore, we find the provided instrumentation acceptable.

3.3.24 Letdown Flow-Out

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The applicant's instrumentation is Category 3. The applicant states that the chemical and volume control system letdown isolates post-accident. This automatic isolation occurs with a safety injection signal, a low pressurizer level signal, or a phase A isolation signal. Thus, there is no letdown flow in post-accident operations.

As an accident signal isolates letdown, and no letdown flow occurs post-accident, we find the provided instrumentation acceptable.

3.3.25 Volume Control Tank Level

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The applicant's instrumentation is Category 3. The chemical and volume control system automatically isolates with a safety injection signal.

The regulatory guide recommends a range from the top to the bottom of the tank. The span monitored is actually 70 inches. The tank is longer than this. The applicant does not monitor ten inches of the cylindrical portion of the tank.

The range supplied covers 7/8 of the straight cylindrical shell. The instrument does not monitor the hemispherical ends of the tank where the volume to level ratio is not linear. Based on this, and because this system automatically isolates for accident response, we find the provided instrumentation acceptable.

3.3.26 Component Cooling Water Temperature to Engineered Safety Features System

Regulatory Guide 1.97 recommends a range of 32°F to 200°F for this variable. The applicant's instrumentation has a range of 50°F to 150°F.

The applicant states that the lowest temperature for this system is 60°F, 10°F above the minimum limit of the range. The applicant also states that the highest temperature expected is 120°F, 30°F less than the maximum limit of the range. Based on the applicant's justification, we find the 50°F to 150°F range acceptable.

3.3.27 High Level Radioactive Liquid Tank Level

Regulatory Guide 1.97 recommends instrumentation for this variable with a range from the bottom to the top of the tank. The applicant monitors from 11 inches to 133 inches (above the tank bottom) in the tritiated drain collector tank.

Each end of the tank has an unmonitored volume of about 1000 gallons. This is out of a total volume of 24,700 gallons. Thus, the instrumentation for the tank volume monitors 92 percent of the tank volume. We find this deviation in range minor and acceptable.

3.3.28 Radioactive Gas Holdup Tank Pressure

Regulatory Guide 1.97 recommends instrumentation with a range of zero to 150 percent of design pressure for this variable. This instrumentation measures from zero to 150 psig. 150 psig is the design pressure of these tanks.

Pressure relief valves operate to keep the pressure from exceeding 150 psig. This is the limit of the instrument range. Because the system design keeps the pressure within the instrument range, we find the provided zero to 150 psig range acceptable.

### 3.3.29 Noble Gases -- Auxiliary Building

Regulatory Guide 1.97 recommends instrumentation for this variable to monitor releases to the atmosphere. The regulatory guide recommends a range of $10^{-6}$ microcuries/cc to $10^{3}$ microcuries/cc. The applicant's instrumentation has a range of $10^{-6}$ microcuries/cc to $10^{-2}$ microcuries/cc.

The applicant states that all discharge paths into the auxiliary building automatically isolate before exceeding the instrument range. Upon isolation, the auxiliary building gas treatment system activates. Based on the described system design, we find the range provided acceptable.

### 3.3.30 Particulates and Halogens

Regulatory Guide 1.97 recommends instrumentation with a range of $10^{-3}$ microcuries/cc to $10^{2}$ microcuries/cc for all identified release points. The applicant identified a range of $10^{-9}$ microcuries/cc to $10^{-4}$ microcuries/cc for the auxiliary building instrumentation.

The applicant states that the auxiliary building discharge path automatically isolates before exceeding the range. Upon isolation, the auxiliary building gas treatment system activates. Additional laboratory analysis of collected samples is available. Based on the described system design, we find the provided range acceptable.

### 3.3.31 Radiation Exposure Meters

Revision 2 of Regulatory Guide 1.97 recommended continuous indications at fixed locations for this variable. The applicant is not providing instrumentation for this variable at the Watts Bar Station. Revision 3 of the regulatory guide deletes this variable. Because of this, it is acceptable not to provide instrumentation for this variable.

### 3.3.32 Wind Speed

Revision 2 of Regulatory Guide 1.97 recommends wind speed instrumentation with a range of zero to 67 miles per hour. The applicant has wind speed instrumentation that reads from zero to 50 miles per hour. This conforms to the recommendations of Revision 3 of the regulatory guide and is acceptable.

### 3.3.33 Accident Sampling (Primary Coolant, Containment Air, and Sump)

Regulatory Guide 1.97 recommends sampling and onsite analysis capability for the reactor coolant system, containment sump, emergency core cooling system pump room sumps, other auxiliary building sump liquids, and containment air.

The applicant's post-accident sampling system deviates from the recommendations of the regulatory guide in the following areas.

1. chloride concentration can resolve down to 1 part per million rather than zero.

2. dissolved hydrogen concentration can resolve down to 10 cc/kg rather than zero.

3. dissolved total gas concentration can resolve down to 10 cc/kg rather than zero.

4. boron concentration can resolve down to 50 parts per million rather than zero.

5. hydrogen content of the containment atmosphere can resolve up to 10 percent rather than 30 percent.

6. oxygen content of the containment atmosphere is not available.

The NRC reviewed and approved the applicant's post-accident sampling facility as part of their review of NUREG-0737, Item II.B.3. We find this a good faith attempt to meet NRC requirements (as defined in NUREG-0737, Supplement No. 1, Section 3.7 [Reference 3]) and, therefore, acceptable.

### 3.3.34 Recording of Category 1 Instrumentation

Regulatory Guide 1.97 recommends recording all Category 1 variables. The regulatory guide recommends use of dedicated recorders when direct or immediate trend or transient information is essential for operator information or action. Computer recording, if continuously updated, or intermittent devices, such as data loggers or scanning recorders, are acceptable if no significant loss of transient response information occurs with such a device.

The applicant states all Category 1 variables will have at least one channel recorded by the emergency response facility data system. The input up to the emergency response facility data system input buffer is Category 1. The applicant uses qualified isolation devices in this application. The applicant also states the following Category 1 variables will have at least one channel recorded by an isolated non-divisional Category 2 recorder. These recorders are in a mild environment.

- reactor coolant system hot leg water temperature
- reactor coolant system cold leg water temperature
- reactor coolant system wide-range pressure
- steam generator pressure
- reactor coolant system pressurizer level
- steam generator wide-range level
- steam generator narrow-range level
- neutron flux
- containment narrow-range pressure

- containment wide-range pressure
- containment radiation
- main steamline radiation
- refueling water storage tank level
- auxiliary building passive sump level

We find the applicant's compliment of computer recording and dedicated recorders acceptable.

### 3.3.35 Isolation Devices for Category 2 Instrumentation

Regulatory Guide 1.97 recommends Category 2 signals sent to other devices use Category 2 isolation devices. The applicant lists this requirement in Reference 6 as not applicable.

The applicant clarifies their use of isolation devices in Reference 9. All Category 1 instruments meet the isolation requirements of Regulatory Guide 1.97. All Category 2, Class 1E, instruments either use Class 1E isolators or the applicant has scheduled upgrades of the secondary instruments to Category 2 requirements. Category 2, non-Class 1E instruments have isolation as follows. The plant annunciator has isolation from this instrumentation by optical isolators. Digital inputs to the emergency response facility data system and the plant computer use optical isolation. Analog inputs to the emergency response facility data system and plant computer enter the input buffers via a resistor network. The input buffers provide the isolation by employing transformer isolating circuits. With these circuits isolated as described, the applicant has isolated the Category 2 instrumentation in conformance to Regulatory Guide 1.97.

### 3.3.36 Display Locations

Regulatory Guide 1.97 recommends post-accident plant variables for the control room operating personnel. The applicant listed several variables in Reference 6 with no indication in the control room. In Reference 9, the applicant shows that, except for 120-Vac inverter voltage and current instrumentation, the variables listed in Reference 6 are now available in the control room via the emergency response facility data system.

The inverter voltage and current displays are assessable and near the control room. A trouble alarm alerts the operator to either abnormal voltage, abnormal current, or other inverter problems.

Section 6.2 of Supplement No. 1 to NUREG-0737 (Reference 3) allows displays in places other than the control room. Therefore, we find the instrumentation display location acceptable.

3.3.37 Core Exit Temperature

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. Category 1 requirements include separation and redundancy of channels. The applicant states the cable bundling at the common reactor vessel refueling cavity wall penetration does not fully follow the recommended separation.

There are 16 thermocouples designated for two channels of post-accident monitoring instrumentation. The core thermocouple system includes additional thermocouples beyond these two channels. The stainless steel mineral-insulated thermocouple cables are in close proximity to control rod mechanisms and rod position indicator stacks. This lessens the possibility of all 16 thermocouples being damaged simultaneously by a single event. Due to the low voltages of a thermocouple signal, a cable fault will not propagate to adjacent thermocouple cables.

Based on this mitigating design, we find this deviation from the full separation recommended acceptable.

## 4. CONCLUSIONS

Based on our review, we find that the applicant either conforms to or has adequate justification for deviating from the guidance of Regulatory Guide 1.97 for each variable.

## 5. REFERENCES


1. Letter, NRC (D. G. Eisenhut) to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement No. 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

2. <u>Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During the Following an Accident</u>, Regulatory Guide 1.97, Revision 2, NRC, Office of Standards Development, December 1980.

3. <u>Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability</u>, NUREG-0737, Supplement No. 1, NRC, Office of Nuclear Reactor Regulation, January 1983.

4. Letter, Tennessee Valley Authority (L. M. Mills) to NRC (E. Adensam), January 30, 1984.

5. Letter, Tennessee Valley Authority (R. H. Shell) to NRC (E. Adensam), April 16, 1985.

6. Letter, Tennessee Valley Authority (E. G. Wallace) to NRC, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Conformance to Regulatory Guide (RG) 1.97, Revision 2 (TAC No.63645)," August 31, 1990.

7. Letter, Tennessee Valley Authority (E. G. Wallace) to NRC, "Watts Bar Nuclear Plant (WBN) - NUREG 0737 Item II.F.1 - Additional Accident Monitoring Instrumentation (TAC 63645)," October 11, 1990.

8. Letter, Tennessee Valley Authority (E. G. Wallace) to NRC, "Watts Bar Nuclear Plant (WBN) - NUREG 0737 Item II.F.1(2), Sampling and Analysis of Plant Effluents - License Condition (LC) 6b," January 3, 1991.

9. Letter, Tennessee Valley Authority (J. H. Garrity) to NRC, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Emergency Response Capability, Regulatory Guide 1.97, Revision 2 - Request for Additional Information Response (TAC Nos. 77550 and 77551)," October 29, 1991.

10. <u>Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident</u>, Regulatory Guide 1.97, Revision 3, NRC, Office of Nuclear Regulatory Research, May 1983.

APPENDIX W

SAFETY EVALUATION REPORT: COMPLIANCE WITH ATWS RULE 10 CFR 50.62*

*Originally issued by letter, S. C. Black to O. D. Kingsley (TVA), December 28, 1989.

ENCLOSURE

# SAFETY EVALUATION REPORT
# WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2
# COMPLIANCE WITH ATWS RULE 10 CFR 50.62
# DOCKET NO. 50-390/391

## 1.0 INTRODUCTION

On July 26, 1984, the Code of Federal Regulations (CFR) was amended to include Section 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants" (known as the ATWS Rule). The requirements of Section 10 CFR 50.62 apply to all commercial light-water-cooled nuclear power plants.

An ATWS is an anticipated operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power) that is accompanied by a failure of the Reactor Trip System (RTS) to shut down the reactor. The ATWS Rule requires specific improvements in the design and operation of commercial nuclear power facilities to reduce the probability of failure to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event.

Paragraph (c)(1) of 10 CFR 50.62 specifies the basic ATWS mitigation system requirements for Westinghouse plants. Equipment, diverse from the RTS, is required to initiate the auxiliary feedwater (AFW) system and a turbine trip for ATWS events. In response to paragraph (c)(1), the Westinghouse Owners Group (WOG) developed a set of conceptual ATWS mitigating system actuation circuitry (AMSAC) designs generic to Westinghouse plants. WOG issued Westinghouse Topical Report WCAP-10858, "AMSAC Generic Design Package," which provided information on the various Westinghouse designs.

The staff reviewed WCAP-10858 and issued a safety evaluation of the subject topical report on July 7, 1986 (Ref. 1). In this safety evaluation, the staff concluded that the generic designs presented in WCAP-10858 adequately meet the requirements of 10 CFR 50.62. The approved version of the WCAP is labeled WCAP-10858-P-A.

During the course of the staff's review of the proposed AMSAC design, the WOG issued Addendum 1 to WCAP-10858-P-A by letter dated February 26, 1987 (Ref. 2). This Addendum changed the setpoint of the C-20 AMSAC permissive signal from 70% reactor power to 40% power. On August 3, 1987, the WOG issued Revision 1 to WCAP-10858-P-A (Ref. 3), which incorporated Addendum 1 changes and provided details on changes associated with a new variable timer and the C-20 time delay. For those plants selecting either the feedwater flow or the feedwater pump/valve status logic option, a variable delay timer is to be incorporated into the AMSAC actuation logics. The variable time delay will be inverse to reactor power and will approximate the time that the steam generator takes to boil down to the low-low level setpoint upon a loss of main feedwater (MFW) from any given reactor power level between 40% and 100% power. The time delay on the C-20 permissive signals for all logics will be lengthened to incorporate the maximum time that the steam generator takes to boil down to the low-low level setpoint upon a loss-of MFW with the reactor operating at 40% power. The staff considers the Revision 1 changes to be acceptable.

Paragraph (c)(6) of the ATWS Rule requires that detailed information to demonstrate compliance with the requirements be submitted to the Director, Office of Nuclear Reactor Regulation (NRR). In accordance with paragraph (c)(6) of the ATWS Rule, Tennessee Valley Authority (TVA) (licensee) provided information by letter dated February 28, 1989 (Ref. 4). The letter forwarded the detailed description of the ATWS mitigating system actuation circuitry proposed for installation at the Watts Bar Nuclear Plant, Units 1 and 2 (WBN).

The staff held a conference call with the licensee to discuss the AMSAC design. As a result of the conference call, the licensee responded to the staff concerns by letter dated August 30, 1989 (Ref. 5).

## 2.0. REVIEW CRITERIA

The systems and equipment required by 10 CFR 50.62 do not have to meet all of the stringent requirements normally applied to safety-related equipment. However, the equipment required by the ATWS Rule should be of sufficient quality and reliability to perform its intended function while minimizing the potential for transients that may challenge the safety systems, e.g., inadvertent scrams.

The following review criteria were used to evaluate the licensee's submittals:

1. The ATWS Rule, 10 CFR 50.62.

2. "Considerations Regarding Systems and Equipment Criteria," published in the Federal Register, Volume 49, No 124, dated June 26, 1984.

3. Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related."

4. Safety Evaluation of WCAP-10858 (Ref. 1).

5. WCAP-10858-P-A, Revision 1 (Ref. 3).

## 3.0 DISCUSSION AND EVALUATION

To determine that conditions indicative of an ATWS event are present, the licensee has elected to implement the WCAP-10858-P-A AMSAC design associated with monitoring the steam generator water level and activating the AMSAC when the water level is below the low level setpoint established for the reactor protection system (RPS). The AMSAC at WBN will contain two low-low level setpoints. The first setpoint will be set at 12 percent of steam generator level when the reactor power is between 40 and 80 percent power as indicated by the turbine first stage impulse pressure. The second steam generator low-low level setpoint will be set at 25 percent of steam generator level when reactor power is above 80 percent. Also, the licensee will implement the new time delay (as described in the introduction section) associated with the C-20 permissive. However, the calculation to establish the time setpoint for this delay is not completed at this time and that Watts Bar committed to produce the setpoint for the AMSAC time delay during the design phase and to complete it before startup.

Many details and interfaces associated with the implementation of the final AMSAC design are of a plant-specific nature. In its safety evaluation of WCAP-10858, the staff identified 14 key elements that require resolution for each plant design. The following paragraphs provide a discussion on the licensee's compliance with respect to each of the plant-specific elements.

1. Diversity

The plant design should include adequate diversity between the AMSAC equipment and the existing Reactor Protection System (RPS) equipment. Reasonable equipment diversity, to the extent practicable, is required to minimize the potential for common-cause failures.

The licensee has provided information to confirm that the AMSAC logic circuits will be diverse from the logic circuits of the RPS in the areas of design, equipment, and manufacturing. Where similar types of components are used, such as relays, the AMSAC will utilize a relay of a different make and manufacturer.

2. Logic Power Supplies

Logic power supplies need not be Class 1E, but must be capable of performing the required design function upon a loss of offsite power. The logic power must come from a power source that is independent from the RPS power supplies.

The licensee has provided information verifying that the logic power supplies selected for the Watts Bar AMSAC logic circuits will provide the maximum available independence from the RPS power supplies. The AMSAC power will be provided by a 120 VAC preferred power board which will be independent of the RPS and capable of operating upon a loss of offsite power. This power supply is non-safety related, noninterruptable, and battery backed.

3. Safety-Related Interface

The implementation of the ATWS Rule shall be such that the existing RPS continues to meet all applicable safety criteria.

The proposed AMSAC design interfaces at its input with the existing Class 1E circuits of the steam generator level instrumentation. At its output, the AMSAC will interface with the Class 1E circuits of the plant's AFW pumps. Connections with these Class 1E circuits will be made through the use of approved Class 1E isolation devices. The licensee has confirmed to the staff that the existing safety-related criteria that are in effect at the plant will continue to be met after the implementation of AMSAC (i.e., the RPS will continue to perform its safety functions without interference from AMSAC). Refer to Item 9 for further discussion of this issue.

4. Quality Assurance

The licensee is required to provide information regarding compliance with Generic Letter (GL) 85-06, "Quality Assurance for ATWS Equipment That is Not Safety Related."

The licensee stated that the quality assurance practices at Watts Bar, as applicable to the nonsafety-related AMSAC equipment, will comply, as a minimum, with the quality assurance guidance as set forth in GL 85-06.

5. Maintenance Bypasses

Information showing how maintenance at power is accomplished should be provided. In addition, maintenance bypass indications should be incorporated into the continuous indication of bypass status in the control room.

The licensee provided information showing how maintenance will be accomplished at power. The staff was informed that maintenance at power will be provided by inhibiting the operation of AMSAC's logic output, which will block the output signal and, thus, prevent it from reaching the final actuation devices. The continuous indication of bypass status will be provided in the main control room through the use of built-in status indication. The licensee further stated that a human-factors evaluation of the subject indication consistent with the plant's control room design process has been performed.

6. Operating Bypasses

The operating bypasses should be indicated continuously in the control room. The independence of the C-20 permissive signal should be addressed.

The AMSAC inhibit/permissive signal (C-20) will be generated by the turbine first stage impulse chamber pressure signals. The AMSAC will be blocked (inhibited) whenever these pressure signals indicate the reactor power is below 40 percent and the AMSAC will be armed (permissive) when the reactor power is at or above 40 percent power. The C-20 signal will be maintained for a period of time within the boundaries as set in WCAP 10858. The time delay setpoint will be calculated during the design phase of the AMSAC.

The C-20 signal will be generated by AMSAC dedicated pressure transmitters and instrumentation and as such will be independent from the RPS. The C-20 status will be indicated in the main control room via the switch module status monitor windows.

7. Means for Bypasses

The means for bypassing shall be accomplished by using a permanently installed, human-factored, bypass switch or similar device. Disallowed methods for bypassing mentioned in the guidance should not be used.

The licensee's response stated that bypassing AMSAC during testing and maintenance will be performed through the use of a permanently installed control switch. The disallowed methods for bypassing such as lifting leads, pulling fuses, blocking relays, or tripping breakers will not be used. The licensee has conducted a human-factors review of the bypass controls consistent with the plant's detailed control room design review process.

8. Manual Initiation

Manual initiation capability of the AMSAC mitigation function must be provided.

The licensee discussed how manual turbine trip and auxiliary feedwater actuation are accomplished by the operator. In summary, the operator can use existing manual controls located in the control room to perform a turbine trip and to start auxiliary feedwater flow should it be necessary. Thus, no additional manual initiation capability will be required as a result of installing the AMSAC equipment.

9. Electrical Independence From Existing Reactor Protection System

Independence is required from the sensor output to the final actuation device, at which point nonsafety-related circuits must be isolated from safety-related circuits by qualified Class 1E isolators.

The licensee discussed how electrical independence is to be achieved. The proposed design requires isolation between the non-Class 1E AMSAC and the Class 1E circuits associated with the steam generator level input signals and the AMSAC output signals to the AFW pumps.

The licensee had informed the staff that the required isolation will be achieved using electrical isolation devices that have been qualified and tested to Class 1E electrical equipment requirements. In addition, the licensee will perform calculations during the design phase to determine that the AMSAC will not present a greater challenge to the isolation devices than that to which they were qualified. The implementation of the AMSAC design will be consistent with the electrical separation criteria established for the plant.

10. Physical Separation from Existing Reactor Protection System

The implementation of the ATWS mitigating system must be such that the separation criteria applied to the existing RPS are not violated.

The licensee stated that the AMSAC circuitry will be physically separated from the RPS circuitry. In addition, the ATWS equipment cabinets will be located so that there will be no interaction with the protection system cabinets. The licensee also stated that the RPS design will continue to meet (subsequent to the implementation of AMSAC) the physical separation criteria originally established for the Watts Bar Plant as stated in the plant's FSAR.

11. <u>Environmental Qualification</u>

The plant-specific submittal should address the environmental qualification of ATWS equipment for anticipated operational occurrences.

The licensee stated that AMSAC mitigation equipment will be located in areas of the plant that are considered to be a mild environment. The licensee also stated that the equipment will be designed to operate in the environment for anticipated operational occurrences that might occur associated with the respective equipment locations.

12. <u>Testability at Power</u>

Measures to test the ATWS mitigating system before installation, as well as periodically, are to be established. Testing may be performed with the system in the bypass mode. Testing from the input sensor through the final actuation device should be performed with the plant shut down.

The licensee has stated that the AMSAC will be tested end-to-end at each refueling outage. The AMSAC is capable of being tested while the plant is at power. The at power test frequency will be based upon the manufacturer's recommendations.

It is the staff's understanding that the licensee will conduct a human-factors review of the controls and indications used for testing purposes that is consistent with the plant's detailed control room design review process.

13. <u>Completion of Mitigative Action</u>

The licensee is required to verify that (1) the protective action, once initiated, goes to completion and (2) the subsequent return to operation requires deliberate operator action.

The licensee responded that the AMSAC system design will be such that the AMSAC action will be consistent with the circuitry of the auxiliary feedwater and turbine trip control systems. Once initiated, the design will ensure that protective action goes to completion. Following completion of the mitigative action, deliberate operator action will then be required to return the actuated devices to normal operation.

14. Technical Specifications

The plant-specific submittal should address technical specification requirements for AMSAC.

The licensee stated that no technical specification action is proposed with respect to the AMSAC and that normal administrative controls are sufficient to ensure AMSAC operability.

The equipment required by the ATWS Rule to reduce the risk associated with an ATWS event must be designed to perform its functions in a reliable manner. A method acceptable to the staff for demonstrating that the equipment satisfies the reliability requirements of the ATWS Rule is to provide limiting conditions for operation and surveillance requirements in the technical specifications.

In its Interim Commission Policy Statement of Technical Specification Improvements for Nuclear Power Plants [52 Federal Register 3788, February 6, 1987], the Commission established a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in technical specifications. The staff is presently reviewing ATWS requirements to criteria in this Policy Statement to determine whether and to what extent technical specifications are appropriate. Accordingly, this aspect of the staff review remains open pending completion of, and subject to the results of, the staff's further review. The staff will provide guidance regarding the technical specification requirements for AMSAC at a later date.

## 4.0 CONCLUSION

The staff concludes, based on the above discussion and pending resolution of the technical specification issue, that the AMSAC design proposed by the Tennessee Valley Authority for the Watts Bar Nuclear Plant, Units 1 and 2, is acceptable and is in compliance with the ATWS Rule, 10 CFR 50.62, paragraph (c)(1).

Even though the staff review regarding the use of technical specifications for ATWS requirements is incomplete, the licensee should continue with the scheduled installation and implementation (planned operation) of the ATWS design utilizing administrative controlled procedures.

## 5.0 REFERENCES

1. Letter, C. E. Rossi (NRC) to L. D. Butterfield (WOG), "Acceptance for Referencing of Licensing Topical Report," July 7, 1986.

2. Letter, R. A. Newton (WOG) to J. Lyons (NRC), "Westinghouse Owners Group Addendum 1 to WCAP-10858-P-A and WCAP-11233-A: AMSAC Generic Design Package," February 26, 1987.

3. Letter, R. A. Newton (WOG) to J. Lyons (NRC), "Westinghouse Owners Group Transmittal of Topical Report, WCAP-10858-P-A and WCAP-11233-A, Revision 1, AMSAC Generic Design Package," August 3, 1987.

4. Letter, R. Gridley (TVA) to U.S. NRC, "Watts Bar Nuclear Plant (WBN) - Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) - Response to NRC's Request for Additional Information," February 28, 1989.

5. Letter, R. H. Shell (TVA) to U.S. NRC, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Circuitry (AMSAC) - Supplemental Response to NRC's Request for Additional Information," August 30, 1989.

APPENDIX X

SUPPLEMENTAL SAFETY EVALUATION*
REVISION 5 OF THE QA RECORDS CORRECTIVE ACTION PROGRAM

*Originally issued by letter, P. S. Tam to M. O. Medford, dated June 9, 1992.

# SUPPLEMENTAL SAFETY EVALUATION

# BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# TENNESSEE VALLEY AUTHORITY

# WATTS BAR NUCLEAR PLANT, UNIT 1

# REVISION 5 OF THE QA RECORDS CORRECTIVE ACTION PROGRAM

# DOCKET NO. 50-390

## INTRODUCTION

In a letter to TVA dated December 8, 1989, the staff reported that Watts Bar "has established acceptable program guidelines for resolution of the QA records issues" within the scope of the corrective action program (CAP). That evaluation was based on the staff's review of Revision 3 of the CAP plan and Section 2.13, "QA Records", of the Watts Bar Nuclear Performance Plan, Volume 4. Subsequently, the staff performed an inspection (Inspection Report 50-390/90-08, dated September 13, 1990) and expressed a concern that the implementation of the CAP may not provide an acceptable Q-list and all records required for licensing (letter, S. D. Ebneter to O. D. Kingsley dated October 30, 1990). As a result, a number of TVA-NRC meetings took place, and a number of letters addressing the staff's concerns were exchanged.

## DISCUSSION AND EVALUATION

After a December 12, 1990 TVA-NRC meeting in the NRC Region II office, TVA documented the information presented at that meeting in a letter to the NRC dated January 28, 1991. Enclosure 1 to that letter described an Additional Systematic Records Review (ASRR). The ASRR was designed to provide additional confirmation of the adequacy of the QA records for Watts Bar Unit 1. The staff commented on the proposed ASRR by letter to TVA dated March 20, 1991, and TVA responded by letter dated May 10, 1991. In a letter dated July 2, 1991, TVA clarified that it would not use any of the data from previous reviews but would use only the ASRR results in performing the planned analysis. By letter dated October 16, 1991, TVA responded to NRC questions of August 30, 1991, and indicated that a modified CAP plan for Watts Bar QA records would be submitted at a later date. TVA's letter dated December 6, 1991, submitted Revision 4 of the CAP which incorporated the ASRR as an attachment. Staff questions regarding Revision 4 of the CAP plan (including the ASRR) were discussed in a TVA-NRC meeting at NRC headquarters on January 27, 1992 (summary dated January 30, 1992), and formally responded to by TVA letter dated February 14, 1992. The QA records CAP was again discussed at a TVA-NRC meeting in Region II offices on March 9, 1992. An NRC staff position was issued to TVA by letter dated April 10, 1992, and discussed during the NRC's inspection of Watts Bar (Inspection Report 50-390/92-10) during the week of May 4, 1992. By letter dated May 15, 1992, TVA submitted Revision 5 of the CAP, which was to be responsive to the staff position.

The staff has reviewed Revision 5 of the CAP. Revision 5 implies that when sampling of QA records does not show an acceptable population, sampling will be stopped and the population represented by the sample will be rejected by TVA on a case-by-case basis using engineering judgment. Therefore, the staff finds that Revision 5 is responsive to the staff position stated above. The staff concludes that Revision 5 of the CAP, when properly implemented, will provide reasonable assurance of the availability of sufficient QA records for issuance of an operating license to Watts Bar Unit 1. The staff will continue to perform inspections/audits to ensure that this CAP has been adequately implemented, QA record problems within the scope of this CAP have been corrected, and methods have been established for recurrence control. In addition, these inspections or audits will also evaluate whether there are other QA records problems at Watts Bar and whether the records necessary to support fuel loading will be available at that time.

Principal contributors: John G. Spraul and Lee R. Abramson

Dated: June 9, 1992

# APPENDIX Y

## AUDIT REPORT
## CORRECTIVE ACTION PROGRAM ON CABLE ISSUES

AUDIT REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CORRECTIVE ACTION PROGRAM ON CABLE ISSUES

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1

DOCKET NO. 50-390

## 1.0 OBJECTIVE

During the week of March 2 thru 6, 1992, a team consisting of the NRC staff and two consultants, J. B. Gardner and W. Thue audited TVA's documentation, and performed plant walkdowns to assess the adequacy of actions taken by TVA to resolve significant cable installation issues that remained open from previous audits at Watts Bar. The NRC staff had previously issued a safety evaluation (SE) on the corrective action program provided by TVA for these issues. The safety evaluation was issued by letter on April 25, 1991, and as Appendix P to Supplement 7 of the Watts Bar Safety Evaluation Report (NUREG-0847, Supp. 7). Previous staff audits were conducted during the week of December 10 - 14, 1990, and January 7 - 11, 1991 (see publicly available trip report by H. Garg dated June 3, 1991).

The issues addressed in this audit are:

1. Mid-route flex conduits
2. Cable jamming
3. Brand Rex cable anomalies
4. Mechanical damage
5. Cable ampacity
6. Cable bend radius
7. cable support in vertical trays and conduits

## 2.0 Discussion

### 2.1 Mid-route flex conduits

TVA documented its evaluation of cables routed in conduits containing mid-route flexible segments in calculation WBP EVAR 9103005 (RIM NO. B 18 910807 900). The evaluation was conducted on hi-pot (high-potential) tested cable in conduits with flexible segments. Sixteen out of 40 conduits that were hi-pot tested for the pullby issue contained mid-route flexible segments and those conduits were used to address this issue. However, two conduits did not contain water in the flexible segment. Therefore, only 14 conduits, containing 121 cables with a total of 429 conductors have been wet tested. An additional conduit with mid-route flexible segment was examined by the University of Connecticut (U-Conn). This flexible segment was part of the cable run which contained damaged cables found in July 1989. The U-Conn report did not attribute any damage of these cables to the mid-route flex segment.

The staff agrees with TVA's assessment that mid-route flex conduits have not caused any significant damage to cables that would affect their ability to perform their intended safety function.

2.2 Cable Jamming

In the SE, the staff has conditionally accepted TVA's program to resolve the cable jamming issue. The staff agreed with TVA to use cables that were being removed from conduits for other cable issues to determine acceptability for jamming, as long as the sample size evaluted contained sufficient numbers of cables with the potential for jamming in order to make a statistical inference of the integrity of the installation as it relates to the jamming issue.

TVA calculation WBP EVAR 900 8003 Rev. 1, attachment 6 identified 76 conduits with a potential jamming problem. TVA is replacing cables from 24 out of these 76 conduits because of some other concerns and will inspect those cables for jamming damage. The staff agrees that cables from 24 out of the 76 conduits provides a sufficient sample to make the determination regarding acceptability of cables for the cable jamming issue. The NRC resident inspector will follow this issue and inspect these cables for any damage caused by jamming. If no damage is found then this issue will be closed by the resident inspector.

2.3 Brand Rex Cable anomalies

During the in-situ hi-pot testing, five cables manufactured by Brand Rex on contract 80 K6-825419 failed the test between 2500 and 4900 V dc. Another Brand Rex cable purchased on the same contract failed the test at Browns Ferry. There were no failures on cables supplied by Brand Rex on other contracts. The cables that failed were removed from the conduit and visually inspected. Since no evidence of pullby damage was found, the cables were sent to U-Conn for further evaluation. By letter dated November 21, 1991, TVA submitted the U-Conn report to the staff. U-Conn performed electrical tests, visual and microscopic examinations, physical measurements, chemical analyses, and morphological examinations on these cables. Based on the tests and examinations, U-Conn concluded that the cables failed due to anomalous large inorganic insulation components such as antimony, titanium and silicon.

In order to determine whether similar problems existed for other Brand Rex cables supplied under different contracts, U-Conn tested cables supplied under contract 79K5-824279-1 and 82k5-830040-2. Based on this testing, U-Conn determined the inorganic particles found in the failed cables at the fault location were present in these cables also, but the sizes of these particles were uniformly distributed and were smaller compared to those of the failed cables.

TVA performed environmental qualification (EQ) tests on a sample of 40 Brand Rex cables to determine that installed cables similar to the ones that failed the test will meet their performance requirements. The sample was selected from cable reels and conduits which contained the failed cables. All cables passed the EQ test. After the EQ test, TVA performed a hi-pot test on these cables, and four of the forty cables failed the test. The failed cables were sent to U-Conn for examination and testing, and U-Conn concluded that large inorganic particles were indeed present at the fault sites. The sequence of EQ test followed by hi-pot test demonstrates that even with the large inorganic particles present, these cables were able to perform their intended function. Cable failure at locations of inorganic particle inclusion during the hi-pot test demonstrated that the reduced insulation was sufficient for the cables to meet their performance requirements since these cables had already passed the EQ test.

The staff has reviewed the EQ test report (E13 911004 265) and determined that, in general the test report is acceptable but noticed an inconsistency in the data log where the "prepared by" space was signed after the "witnessed and understood by" space was signed, implying that the person has not witnessed the entry. Also in some cases the space "witnessed and understood by" is signed four months after the space "prepared by" is signed. It simply implies that the person has reviewed the data entry at a later date. The staff asked TVA to verify whether that meets their QA program and TVA issued a Problem Evaluation Report (PER) WBPER 920067. The resident inspector will follow the closure of this issue.

Based on the above evaluation the staff concludes that TVA has adequately addressed the issue of Brand Rex cables pending resolution of PER WBPER 920067.

### 2.4 Mechanical Damage

During hi-pot testing, many cables failed the test because of mechanical damage to the cables. These cables were sent to U-Conn to confirm the cause of the damage. In most cases, except for two cables, U-Conn was able to determine the root cause of damage mechanism. The root cause of the damage included kink, splice, damage caused by sharp object etc. In their letter of August 22, 1991, TVA attributed this damage as random events without basis. During the audit, TVA cited that PERs, WBP 900 400 PER Rev. 6 , WBP 900 428 PER Rev. 4, WBP 900 548 PER Rev. 0 and WBP 900 550 PER Rev. 1 document the basis for the random and isolated cases of mechanical damage. However, these PERs did not contain proper justification. A later revision to the PER provided a satisfactory justification for considering the damage mechanism as isolated events with the exception of splices or termination damage. For these damages, TVA issued WBP 900 498 SCA and WBP 900 450 SCA. The NRC resident inspector is following these significant corrective action reports (SCAR). The staff finds that the revision to PER has adequately addressed the question of mechanical damage to cables.

TVA did not document the above cable test failures in a CAQR (condition adverse to quality report) because they made the judgement that availability of redundant systems satisfied the requirements. The NRC staff does not agree with this justification and feels that there should be a time limit for the resolution of a PER for any safety system component failure before it is documented in a CAQR. The NRC resident inspector is following this item.

2.5 Cable Ampacity

The implementation of the TVA ampacity program for class 1E cables was examined in detail during this audit and calculations for cables in trays and conduits, as well as cables in duct bank were reviewed. A plant walkdown was conducted to verify the input data used in the calculation.

2.5.1 Cables in Trays and Conduits

2.5.1.1 Documentation Review

TVA calculation WBPEVAR8909010 documenting the cable ampacity for NV4 and NV5 cables in cable raceways was reviewed during this audit. The review of this calculation indicated that in general TVA has used good engineering judgement, the assumptions were reasonable and generally conservative, and the results are acceptable. The documentation of assumptions and the methodology was found complete. Instructions for walkdowns included consideration of tray/conduit fill, fire barriers, tray covers, fire wraps etc.

One of the items of concern to the staff which was documented in the SER was the derating for tray covers. TVA has elected to derate by 25% all cables routed in trays with covers longer than 6 feet. This resolves the staff's concern.

2.5.1.2 Plant Walkdowns

The adequacy of the walkdown results was audited and the following five cable runs were selected at random to confirm the results of TVA's walkdown data.

1PL 4900 A 1PL 4901 B 2PL 4901 B
1PL 4735 S 1PL 4736S

The selection was based on the relatively high ampacity circuits which were very close to the allowable ampacity of the cables. Since the replacement of some cables had begun on most of these runs, some of the tray covers were removed from the tray and were not reinstalled as shown in the original walkdowns. However, the evidence of their original placement was obvious.

All of the runs were properly evaluated in the original walkdown except for cable 1PL 4901 B. A 2-foot curved tray cover was in place but was not shown on the original walkdown sheet of March 24, 1989. Since another raised tray cover of 6 ft. 6 inches was located just below the missing 2-foot section of tray cover, TVA had allowed 25% derating for the tray cover and the missing section did not affect the cable ampacity calculation. However, it raised the concern about the adequacy of TVA walkdowns.

TVA had previously issued a SCAR WBP900050SCA on February 2, 1990, as a result of TVA's independent verification of the tray/conduit configurations documented on the walkdowns. TVA's nuclear engineering group has reviewed the discrepancies found during the independent verification study and concluded that in all cases the error was insignificant to the engineering evaluation. The staff was still concerned, since such discrepancies can be significant only for the cases where the tray covers are documented as less than or equal to 6 ft. Therefore, the staff asked TVA to perform walkdown of the 11 tray segments which have tray covers close to 6 ft. The NRC resident inspector will follow this item.

### 2.5.2 Underground Duct Bank

TVA calculation WBP EVAR 9003002 covers the ampacity of cables in duct banks. For this audit, the calculation for the cables in duct banks between the auxiliary building and the intake pumping station was reviewed.

Most of the assumptions used by TVA are very conservative and will result in a lower ampacity value for cables than would have been obtained by more realistic assumptions that are now used by industry. Examples of these conservative assumptions are:

a. Thermal resistivity of concrete
b. Specific inductive capacitance (SIC) value of insulation
c. Shield losses
d. Thermal resistivity of insulation.

TVA has chosen a thermal resistivity of the soil surrounding the duct bank as Rho of 90. IEEE-S135, originally AIEE/IPCEA "Black Books," on ampacity of cables was derived from the Neher-McGrath paper of 1957. Tables in these books were calculated using conditions that affect cable ampacities such as soil thermal resistivity. Three values of Rho have been used in the Black Books: Rho 60, 90 and 120. The book also suggests that when the earth thermal resistivity is not known a value of 90 Rho be used.

The unusual configuration of the duct bank at WBN is caused by the large quantity of cables in the duct bank and is further complicated by the two to six large pipes for intake and circulating water near the duct bank. Adjacent to the turbine building and near MH-2 (manhole-2) a duct bank, with 72 conduits, is located over two 12-ft diameter circulating water pipes. The duct bank is 10 ft wide by 5 ft 1 inch high and the heat loss from the cables is given as 35.1 Watts/ft. The heat loss from each of the two pipes is 27.3 watts/ft in the summer, and 47.4 watts/ft in the winter. This results in a worst-case situation with combined heat loss of 129.9 watts/ft from the duct bank and the two pipes. Experience around the world has shown that soil dryout can occur with heat loss as little as 30 watts/ft for cables in duct bank of less than one foot in diameter.

The soil around WBN is clay and clay soils, which, when heated by a heat source of over 100 watts/ft, are known to lose almost all of their moisture and their thermal resistivity can climb to about 300 Rho. When clay contains the amount of moisture that is found in a normal environment, a thermal resistivity of 90 Rho is common.

This raises a concern regarding aceptability of the WBN calculation with such large amounts of heat. However, the staff believes that the contribution of water pipes may not truly add that much heat to the duct bank.

Hence, the staff requires that a re-evaluation of the calculations be made by TVA, using input from both the electrical and mechanical groups in order to arrive at a more realistic heat flux contribution from the pipes and the duct bank. If this recalculation demonstrates that the heat flux is no more than 100 watts/ft, then the soil thermal resistivity of 90 Rho will be acceptable to the staff. Otherwise, TVA should verify by test the soil thermal resistivity of 90 Rho for the calculated heat flux. This item will remain open and will be followed by the resident inspector.

2.6 <u>Cable Bend Radius</u>

During the previous audit which was conducted January 7-11, 1991, many items related to cable bend radius were left open by the NRC staff. These items were identified in the trip report by H. Garg dated June 3, 1991 (publicly available document). Other open items were also identified in the staff's safety evaluation. During the present audit, the NRC team reviewed the closure of all open items included in the safety evaluation and the trip report. The following discussion provides details:

2.6.1

<u>Open Item:</u>

The lack of formal guidance regarding the methodology used to take cable bend radius measurements placed into question the integrity of the walkdowns completed to date. Consequently, the staff requested that TVA reinspect a sample of 59 cables previously examined, which were retrained, using the methodology described in formal work procedures. Additionally, since these cables had been accepted during the original bend radius walkdowns, any configuration which fails to meet the minimum bend radius criteria should be reported to the NRC.

<u>EVALUATION:</u>

Review of TVA procedure MAI 3.2 Rev 4. Page 52 of 72, Section H1.0 confirmed that TVA walkdown procedures now contain the proper methodologies for field measurement of cable bend radius. TVA has agreed to perform a walkdown of 59 randomly selected conduits to verify that inadequate prior measurement instructions did not result in under-corrected over-bent cables. This walkdown has not been completed due to the stop work order issued in 1991. The work plan for this effort was reviewed by the staff and appears to address the issue properly. NRC resident inspectors will follow the walkdown of the 59 samples.

2.6.2

Open Item:

Future cable bend radius walkdowns should incorporate provisions for recording the "as found" condition of the cables examined.

Resolution:

Sample copies of the old and revised data sheets used in cable walkdowns were reviewed by the team to determine whether changes that were required were made to record "as found" bend radius information. The team found that the revised forms contain a space for "Approximate Measured Radius". However, this information is only filled when the cables are below the upper bound and above the lower bound. Since TVA is retraining the cables to a value other than that generally accepted by the industry, any trending of future failures will be difficult without the complete database. This information could be obtained by TVA later on for trending and the staff feels that for the present time, TVA has adequately addressed the staff's concern.

2.6.3

Open Item:

NRC review of load cycle and corona test for the medium voltage cables.

Resolution

The Okonite Co. is performing the above test for TVA and as of this audit, the test was not completed. The test plan of Febuary 26, 1992, was reviewed by the staff during this audit. The staff was concerned with the incomplete corona measurement data. To resolve this concern, TVA asked Okonite, to provide XY plots of corona versus voltage levels. All other aspects of the test plan were acceptable to the staff.

2.6.4

Open Item:

NRC review of testing of multiconductor cables to establish lower bound.

Resolution:

During the audit, the final report of the multiconductor cable bend tests of January 24, 1992, was reviewed. This test was conducted at TVA's Central Laboratory. The overall test plan and stated objectives appear to be proper and clear.

During the test, tie wraps were used near the bend to hold the cable in position. This could result in great variation in allowing slippage between individual conductors of a cable, and affect the results much more than the rigid clamped and conductor soldering of cables farther away from the bends.

However, compared to probable conditions in the plant, any clamping effects during the test would provide conservative test results.

Because of the poor quality of photo details, the staff was not able to get a good feeling about the jacket distortions and conductor insulation indentation, especially the flattening of the inner conductor of the largest cable. Also the test program did not consider the length of time the cable had remained in the over bent condition. TVA's response to the staff's concern was that for the samples that were left in the over bent condition longer, before straightening for inspection, the conductor insulation indentation was less severe. Since the cables at WBN have been installed in the plant for a long period of time, the conductor insulation indentation will be less severe than what was observed in the test. However, since no test data is available, the question still remains unresolved on a generic basis. Hence, even though the test results are acceptable for WBN application, they may not have the benefit of broad or long experience to back it up to other cable materials or construction. This item is closed for WBN.

2.6.5

Open Item:

Long-term aging effects on cable bending.

Resolution:

No work has been done so far to test and analyze the age-related effects of lower or upper limit bending of cables in mild and harsh environments. This is a long-term concern. TVA plans to implement a test program at the Bellefonte Nuclear Plant to determine the age-related effects. The NRC staff asked TVA to submit their test plan for staff review as soon as it becomes available (TVA should provide in writing an approximate date on which this submittal would be made).

2.6.6

Open Item:

TVA should contact all the cable manufacturers regarding the acceptability of the TVA test program used to establish lower bound bend radius.

Resolution:

TVA has approached the cable manufacturers on the phone about the technical adequacy of TVA's approach to establish the lower bound for cable bend radius and the response has been favorable to the approach TVA used. However, no written response has been received from the manufacturers. The resident inspectors will follow up this item.

## 2.7 Cable Support in Vertical Tray and Conduit

TVA has made substantial progress in addressing the staff's concerns identified during the previous audit, conducted on January 7-11, 1991. Changes to documents and inspections conducted by TVA have eliminated many of the concerns expressed in prior audits. During the present audit the team focused on the closure of these open items. The number in parentheses in front of the open item below identifies the open item from the June 3, 1991, trip report.

2.7.1

Open Item:

(B.1.a) SWBP analysis ignores the dramatic physical effects on some cable materials when exposed to the high temperature of harsh environments.

Resolution:

The cables which are required to function during an accident condition are qualified to the harsh environment created by the accident. Although the qualification test is done in a configuration which is different from the installed condition (long vertical drop), the staff is of the opinion that as long as the cable is installed in accordance with the acceptable installation procedure, it will perform its function. The staff may initiate a program in the future if trend analysis or other evidence of abnormal behavior of cable becomes available from operating plants. Based on such consideration, this item is considered closed.

2.7.2

Open Item:

(B1.b, B5, G1) The analysis of SWBP is conservative in some respects but ignores the extreme bending radius condition or effect of local indenting of cables at tray sides, or duct, or condulet lips which create "hanging points," not simply the assumed cables passing over a round corner.

TVA has assumed 1/8-inch radius bends at all conduit corners. However past observations of "LB" and "T" condulet has indicated that some condulet's inside corners are sharper than the 1/8" radius. TVA should also justify the Kerite report's applicability to sharp condulet bends.

Resolution:

TVA's assessment of acceptability of sharp bends at the top of vertical runs is included in calculations WBP EVAR 9005001 and 17190.5001 EE(B) -007 and is based substantially on Okonite Co. test data and Kerite Co. data book guidance. The NRC staff believes that the applicability of these references to the plant condition is very questionable and the reason for the staff's belief is as follows:

The Okonite test consisted of a 1/4"-OD rod pressing against a straight sample of insulated wire resting on a flat plate with the failure point being the time when the conductor (not under any tension) made contact with the rod or plate by a complete insulation rupture (a splitting or tearing action). The test time was a few seconds to 11 minutes maximum. Applying such data to actual installed condition where the conductors are in tension and the affected areas usually smaller than that pressed on by the 1/4" rod (applying pressure to cause creeping of conductor though the insulation over many years as a failure mode) seems like an extreme extrapolation of geometry and time. Where large conductors are involved or many cables press an underlying cable against a small diameter bend point, the Okonite geometry is more relevant, although the time factor and mode of failure (quick smashing versus long-term creep) are still far different.

Similarly, the Kerite guidance referenced by TVA was developed from far different geometries than related to this sharp corner issue. As clearly implied from the context, the Kerite guidance was developed from tests of cables passing over standard conduit bends where the inside circumferential curvature of the conduit tends to cradle the cable and there is very little shear or indentation type stresses on the insulation or jacket. Industry experience has shown that there is much more damage inflicted on cables by pulling over small diameter rollers around a bend than over a smooth conduit bend. Hence application of the Kerite guidance to a cable resting on a small diameter corner is an extreme application.

The issue of the proper diameter to assume for conduit lips continues to be troublesome. A sample of 9 to 10 varied sizes of condulets visually inspected indicated that the diameter of the rounded lips of most condulets were close to 1/8". This means the radius would be about 1/16", not the assumed 1/8" for the WBN calculations. The team did not find any sharp protrusions from the LB lips that were found during the original plant walkdown in 1987. Apparently the earlier finding was a random occurrence.

The above issues, while not enough to undermine the overall adequacy of the WBN installation, do not imply the approval of the applicability of the Okonite test and Kerite guidance for extending the vertical support requirements of the NEC. Also, guidance should be provided to the craft to observe any sharp corners in condulets, and measurements should be made to determine the bend radius of the LB condulet lips rather than the assumed value of 1/8" radius.

2.7.3 <u>Open Items:</u>

(B.2, F) The staff's review of TVA calculation WBP EVAR 9005001 and WBP EVA 8005001 and WBP EVA 8007011 was not complete with regard to corrective actions taken by TVA.

Also TVA construction Standard G-38 Rev. 10, provides seven different options for support without any justification and guidance to the site construction personnel.

Resolution:

Review of calculation WBPEVAR9005001R1 indicated proper derivation of vertical support practices other than those discussed in Section 2.7.1. Review of WBNEVAR9007011 and WBNEVAR8907010 and related DCNs have indicated an adequate analysis of walkdown reported conditions to determine the location for added supports. The evident difficulty in adding support at these locations and leaving the support choice to site construction engineering decision adds concerns for the issue as noted below.

The large variety of methods for support options accepted by TVA and the apparent flexibility given to construction engineering to choose between them, without much guidance in G-38, raises concerns as to just how they might be applied and will they still be proper and adequate. The staff questioned the applicability of tie wraps or wires used as supporting methods. TVA stated that these methods are generally used in junction boxes. The staff feels that this kind of support will either be inadequate or could cause cable damage. Also applying Raychem NJRS wraps around sleeves over a sharp corner is not a method of support, because they will not limit the longitudinal movement of the cable. Simply padding a corner to increase the effective radius only provides some protection against mechanical and electrical failure probability but is not an effective vertical support. The staff left this as an open item with TVA to provide justification with specific details, or delete them as a support mechanism.

The use of fire stop material to act as a vertical support was previously accepted by the staff for RTV rubber foam when used as a fill in conduit body. However, TVA guidance in G-38 implies that any fire stop material could be used as a means for vertical support. TVA should test these materials for holding power and obtain the cable manufacturer's approval for compatibility with the insulation material. In general, to avoid improper installations, any support method should be substantially tested and have the cable manufacturers' acceptance that it will not be mechanically injurious to, and is compatible with, the specific types of cables involved. For example, the use of oZ Gedney wedge type supports with wood or hard polymer inserts will be mechanically injurious to some cables, especially multiconductor cables over a long period of time. Chico A (a portland cement mixture) will leach strong alkalies to attack some polymers if used in a wet or damp locations.

2.7.4

Open Item:

(B4) The definition of the term "near" for condulets near the top of vertical runs requires clarification . (Top of p. 4 of 19) Also see walkdown procedure WP-47.

Resolution:

The issue of support near the top of vertical runs was discussed with TVA during this audit. It was noted that an IEEE working group is in the process of revising the IEEE-690 which will define when a condulet near the top of a vertical run is a concern. In the meantime the definition used by TVA is acceptable.

2.7.5

Open Item:

(D) Review of walkdown procedure WD006 indicated that it was thorough and clearly written. The procedure contains appropriate forms to be used to record walkdown observations and examples of their use. One question not answered by this procedure or other documents reviewed is at what time and point in the screening/walkdown process was the change made to not take credit for the horizontal runs of power cable for their vertical support.

Resolution:

Walkdown procedure WD006 covers all information needed to make proper vertical support decisions. TVA calculations WBPEVAR9005001 and WBPEVAR9007011 were updated to eliminate any credit taken for horizontal sections of power cable runs and these revised criteria had been and will be applied to all conduits regardless of the time the walkdown occurred. Therefore, the date of the walkdown is not an issue and this item is closed.

2.7.6

Open Item:

(G2) One of the options proposed by TVA to resolve the issue of the assumed damaged silicone rubber insulated cables, based on visual inpections, was to hi-pot test the cables and the cables that pass the hi-pot test would be acceptable. However, the staff did not agree with this option unless a large population of cables were tested and a statistical inference could be drawn from the test results.

Resolution:

NRC never agreed to the option of testing to assure the integrity of a cable which is assured to be damaged. However, TVA has not exercised this option and the DCNs M-11131-A and M-15543-A provide instructions to construction personnel that if cables cannot be lifted off the support point with one hand, then Nuclear Construction (NC) should submit a DCN to Nulcear Engineering (NE) to provide design output for replacement of the cables with LOCA qualified cables. At the time when construction was halted, NE had not received any DCN requests for cable replacement indicating that cable tension was not present for the inspected cables. Based on this the staff consider this item closed.

2.7.7

Open Item:

(G3) In order to assure the acceptability of walkdown information regarding conduit geometry, the staff during its next audit will review the walkdown packages #14, 26, and 29.

Resolution:

The audit team performed the review of the walkdown data sheets for the three packages. This review indicated that the geometries, although very complex, were being evaluated in an acceptable manner. Hence the staff considers this item closed.

3.0 Conclusion

Based on the audit, the staff believes that the TVA corrective action program for cable issues is being conducted in a way that would resolve all concerns related to cable installation. The NRC resident inspectors will continue to monitor the implementation of these programs and will follow the closure of the open items identified in this audit report.

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Supplement No. 9 to the Safety Evaluation Report for the application filed by the Tennessee Valley Authority for license to operate Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391, located in Rhea County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation of (1) additional information submitted by the applicant since Supplement No. 8 was issued, and (2) matters that the staff had under review when Supplement No. 8 was issued.

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